Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-128

June 19, 2015

10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Sequoyah Nuclear Plants, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10) - Supplement 2**

References:
1. TVA Letter to NRC, "Sequoyah Nuclear Plants, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10)," dated November 22, 2013. (ADAMS Accession No. ML13329A717)
2. TVA Letter to NRC, "Sequoyah Nuclear Plants, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10) - Supplement 1," dated December 16, 2014. (ADAMS Accession No. ML14350B364)

By letter dated November 22, 2013, Tennessee Valley Authority (TVA) requested a license amendment to revise the current Technical Specifications for Sequoyah Nuclear Plant (SQN), Units 1 and 2, to the Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications described in NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 4.0 (Reference 1).

By letter dated December 16, 2014 (Reference 2), TVA provided a supplement to the ITS license amendment request (LAR). The supplement provided information regarding the revised SQN fuel handling accident radiological consequences analysis using the alternative source term.

The purpose of this letter is to supplement the original ITS LAR (Reference 1). Specifically, this letter complements and revises the original LAR based on responses to NRC staff requests for additional information (RAIs). The RAIs and the associated responses are posted on a publicly available website, the NRC and SQN ITS Conversion Website (http://www.excelservices.com), and are hereby docketed by submittal of this letter.

This supplement contains: 1) proposed changes to the original ITS LAR resulting from TVA responses to the ITS LAR RAIs; 2) proposed changes resulting from TVA self-identified issues discovered during review of the ITS LAR; and 3) docketed submittal of NRC staff RAIs, TVA responses and NRC staff RAI closures posted as of May 31, 2015.

Enclosure 1, "Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10) – Revision 1," provides a revision of Enclosures 2 (Volumes 1, 3, 4, 5, 6, 7, 9, 10, 15 and 16), 5, 6, 8, and 9 of the original ITS LAR with changes annotated on the affected pages. ITS LAR revisions associated with a TVA response to an RAI are identified on the revised page with a red text box indicating the applicable RAI number (e.g., KAB044). ITS LAR revisions associated with a TVA self identified issue are identified on the revised page with a red text box with an SII indicator. Volumes 1, 3, 4, 5, 6, 7, 9, 10, 15 and 16 of Enclosure 2 of the original ITS LAR have been included because there are no open RAIs associated with them.

Additionally, Enclosure 1 contains a revision to Enclosure 8 of the original ITS LAR. Revised Enclosure 8, "Regulatory Commitments," contains two new commitments associated with responses to RAIs MEH-006 and RPG-014. There are thirteen commitments associated with the ITS LAR. The commitments associated with the ITS LAR are only those contained in the revised Enclosure 8.

Enclosure 2, "SQN Self-Identified Issues," provides a list of self-identified issues discovered during review of the ITS LAR (Reference 1). The list provides a brief description of each self-identified issue, the ITS Section affected by the issue, and the page numbers for affected pages.

Enclosure 3, "SQN ITS Conversion RAI Database," contains the NRC staff RAIs and the associated TVA responses. Each RAI/response includes the question asked by the NRC staff reviewer, the TVA response, proposed changes to pages contained in the ITS LAR (Reference 1), any attached supporting documentation, and RAI closure documentation as of May 31, 2015.

The information provided by this supplement to the original ITS LAR does not change the intent or the justification for the requested ITS license amendment. TVA has further determined that this supplement does not affect the basis for concluding that the proposed license amendment does not involve a Significant Hazards Consideration. As such, the 10 CFR 50.92 evaluation provided in the November 22, 2013, ITS LAR remains valid. In addition, the ITS LAR, including this supplement, continues to be exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

The SQN Plant Operations Review Committee has reviewed this supplemental information and determined that operation of SQN in accordance with the Technical Specifications as proposed in the original ITS LAR and this supplement, will not endanger the health and safety of the public.

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosure to the Tennessee State Department of Environment and Conservation.

If there are any questions or if additional information is needed, please contact Mr. Tom Hess at 423-751-3487.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 19th day of June 2015.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures: Enclosure 1 - Sequoyah Nuclear Plants, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10) - Revision 1

Enclosure 2 - SQN Self-Identified Issues

Enclosure 3 - SQN ITS Conversion RAI Database

Enclosure
cc (Enclosure):

- NRC Regional Administrator – Region II
- NRC Senior Resident Inspector – Sequoyah Nuclear Plant
- Director, Division of Radiological Health - Tennessee State Department of Environment and Conservation
- NRC Project Manager - Sequoyah Nuclear Plant

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications Conversion to NUREG-1431, Rev. 4.0 (SQN-TS-11-10) - Revision 1

ENCLOSURE 2

VOLUME 1

SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

APPLICATION OF SELECTION CRITERIA TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

Revision 0

APPLICATION OF SELECTION CRITERIA
TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS

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APPLICATION OF SELECTION CRITERIA TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Westinghouse Owners Group application of the Technical Specification selection criteria on a plant specific basis for the Sequoyah Nuclear Plant (SQN) Unit 1 and Unit 2. The Tennessee Valley Authority (hereinafter TVA) has reviewed the application and confirmed the applicability of the selection criteria to each of the Technical Specifications utilized in report WCAP-11618, "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application" (Reference 1) including Addendum 1, NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988 and as revised in NUREG-1431, Revision 4.0 "Standard Technical Specifications, Westinghouse Plants" (Reference 2) and applied the criteria to each of the current SQN Technical Specifications. Additionally, in accordance with the NRC Final Policy Statement (Reference 3), this confirmation of the application of selection criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to SQN.

APPLICATION OF SELECTION CRITERIA TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

2. SELECTION CRITERIA

TVA has utilized the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 3) to develop the results contained in the attached matrix. PRA insights as used in the Westinghouse Owners Group submittal were utilized, confirmed by TVA, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 3 are as follows:

> Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:
>
> Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.
>
> This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).
>
> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:
>
> Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing design basis accident and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events

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(ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

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

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

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

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

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated design basis accident or transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the design basis accident or transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's design basis accident and transient analyses,

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

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

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

<u>Discussion of Criterion 4</u>: It is the Commission's policy that licensees retain in their Technical Specifications LCOs, Action statements and Surveillance Requirements for the following systems (as applicable), which operating experience and PRA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant and design-specific PRA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PRA or operating

**APPLICATION OF SELECTION CRITERIA
TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2
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experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

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

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

3. PRA INSIGHTS

Introduction and Objectives

Reference 3 includes a statement that NRC expects licensees to utilize any plant specific PRA or risk survey and any available literature on risk insights and PRAs to strengthen the technical bases for these requirements that remain in Technical Specifications and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed as being relocated to other plant controlled documents will be maintained under programs subject to the 10 CFR 50.59 review process. These Relocated Specifications have been compared to a variety of PRA material with two purposes: 1) to identify if a Specification component or topic is addressed by PRA; and 2) if addressed, to judge if the Relocated Specification component or topic is risk-important. The intent of the PRA review was to provide an additional screen to the deterministic criteria. This review was accomplished in the generic Westinghouse Owners Group submittal WCAP-11618 and Addendum 1 to WCAP-11618 (Reference 1). The results of this generic review have been confirmed by TVA for the applicable SQN Specifications to be relocated.

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Assumptions and Approach

The WCAP-11618 evaluation of the risk impact of the Technical Specifications that are relocation candidates was based on the following:

a. It was assumed that any of the Technical Specifications that were to be relocated would be transferred to other documents subject to control by the utility under the 10 CFR 50.59 process.

b. The risk criteria used in determining the disposition of a Technical Specification were the following:

   1. If the Technical Specification contained constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk, it should be retained;

   2. If the Technical Specification included items involved in one of these dominant sequences but had an insignificant impact on the probability or severity of that sequence, it was proposed to be relocated to another controlled document; and

   3. If the Technical Specification was not involved in risk dominant sequences, it was proposed to be relocated to another controlled document.

c. The measures related to risk used in this evaluation were core damage frequency and off-site health effects. These measures were consistent with the Final Policy Statement on Technical Specifications and the Safety Goal and Severe Accident Policy Statements.

d. The criteria used to determine if a sequence was risk dominant was the following: For core damage, any sequence whose frequency was commonly found to be greater than 1×10^{-6} per reactor year was maintained as a possible dominant sequence as a conservative first cut. This was roughly 2% of the total core damage frequency of 5×10^{-5} for typical PRAs. Each specific sequence identified in the screening of the Technical

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Specifications was evaluated against the above conservative criterion to determine if it was risk dominant.

For off-site health effects, any sequence whose frequency of serious radioactive release was commonly found to be greater than 1 x 10^{-7} per reactor year was considered to be a dominant risk sequence for the purposes of WCAP-11618. This criterion was in agreement with the NRC position in the Safety Goal Policy for a goal of 1 X 10^{-6} for a total frequency of severe off-site release, and no greater than 1 X 10^{-7} for an individual sequence.

e. Included in Section 4.0 of WCAP-11618, were two tables (Tables 3 and 4) which contained representative sequences for all identified types of initiating events considered in formal risk assessments for two types of reference plants. Table 3 was representative of a plant with a large dry containment and Table 4 contained the dominant accident sequences for a plant with a subatmospheric containment. These lists were based on industry PRAs and were reviewed for consistency with NRC sponsored PRA programs. The results were found to be consistent.

Systems identified in Tables 3 and 4 of Section 4.0 of WCAP-11618 that contributed significantly to risk as defined in Paragraph d above were listed in Tables 3A, 3B, 4A and 4B of Section 4.0. These identified systems as well as sequences and the risk dominant initiating events from Tables 3 and 4 which were involved in typical dominant core damage and serious release sequences from formal risk assessments were used to screen the requirements of the Technical Specifications reviewed. Those Technical Specifications whose requirements were relevant to these systems, sequences, and initiating events were further evaluated for risk dominance. The remaining Technical Specifications were evaluated on the basis of risk insights from references listed in Section 4.0, Appendix B of WCAP-11618. If the requirements of a Technical Specification were not found to be modeled in any reference and no significant issues were identified from a review of the risk insights, the conclusion was that it did not contain constraints of prime importance to limiting the likelihood or severity of sequences that are commonly found to dominate risk.

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4. RESULTS OF APPLICATION OF SELECTION CRITERIA

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

**APPLICATION OF SELECTION CRITERIA
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5. REFERENCES

1. WCAP-11618 (and Addendum 1), "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program — Phase II Task 5, Criteria Application," November 1987.

2. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4.0, April 2001.

3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
1.0 1.3 1.7 1.30	**DEFINITIONS**	**1.1** 5.5.1 3.6.3 3.6.1/3.6.3 3.6.7/3.6.10	YES	This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria, will remain as definitions in this section of Technical Specifications.
2.0	**SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS**	**2.0**		
2.1	***Safety Limits***	***2.1***		
2.1.1	Reactor Core-The combination of Thermal Power, Pressurizer Pressure and Highest Operating Loop Coolant Tavg Shall not exceed the limits ~~of the COLR~~ shown in Figure 2.1-1	2.1.1	YES	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.

SII

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
2.1.2	Reactor Coolant System Pressure- Reactor Coolant System Pressure shall not exceed 2735	2.1.2	YES	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.
2.2	***Limiting Safety System Settings***			
2.2.1	Reactor Protection System Setpoints	3.3.1	YES-3	The RTS LSSS have been included as part of the RPS instrumentation Specification, which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RTS.

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.0	**LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS - APPLICABILITY**	**3.0**		
3.0.1	Operational Modes	LCO 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.
3.0.2	Noncompliance	LCO 3.0.2	YES	Same as above.
3.0.3	Generic Actions	LCO 3.0.3	YES	Same as above.
3.0.4	Entry into Operational Modes	LCO 3.0.4	YES	Same as above.
3.0.5	Operability Exception	3.8.1	YES	The application of Technical Specification selection criteria is not appropriate. However, this exception to the definition of OPERABILITY has been included as part of the Required Actions in ITS 3.8.1.

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.0.6	Actions Exceptions	LCO 3.0.5	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.
3.0.7	Snubbers	LCO 3.0.8	YES	
4.0.1	Operational Modes	SR 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
4.0.2	Time of Performance	SR 3.0.2	YES	Same as above.
4.0.3	Noncompliance	SR 3.0.3	YES	Same as above.
4.0.4	Entry into Operational Modes	SR 3.0.4	YES	Same as above.
4.0.5	ASME Code Class 1, 2, and 3 Components	5.5.5 5.5.6	YES	This Specification is actually a Surveillance Requirement which has been retained in the Administrative Controls programs for Inservice Testing.
3/4.1	**REACTIVITY CONTROL SYSTEMS**	**3.1**		
3/4.1.1	***Boration Control***			
3.1.1.1	SHUTDOWN MARGIN- Tavg Greater Than 200°F	1.0 3.1.1 3.1.2 3.1.4 3.1.6	YES-2	
3.1.1.2	SHUTDOWN MARGIN- Tavg Less Than or Equal to 200°F	1.0 3.1.1 3.1.4	YES-2	
3.1.1.3	Moderator Temperature Coefficient	3.1.3	YES-2	
3.1.1.4	Minimum Temperature for Criticality	3.4.2	YES-2	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.1.2	***Boration Systems (deleted)***			
3/4.1.3	***Movable Control Assemblies***			
3.1.3.1	Group Height	3.1.4	YES-2	
3.1.3.2	Position Indicator Systems-Operating	3.1.7	YES-2	
3.1.3.4	Rod Drop Time	3.1.4	YES-2	This Specification has been incorporated as a Surveillance Requirement (SR 3.1.4.3) in ITS 3.1.4.
3.1.3.5	Shutdown Rod Insertion Limit	3.1.5	YES-2	
3.1.3.6	Control Rod Insertion Limits	3.1.6	YES-2	
3/4.2	**POWER DISTRIBUTION LIMITS**	**3.2**		
3/4.2.1	***Axial Power Imbalance***			
3.2.1	Axial Flux Difference(AFD)	3.2.3	YES-2	
3/4.2.2	***Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$***			
3.2.2	Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$	3.2.1	YES-2	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[1]
3/4.2.3	***Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$ (X,Y)***			
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$(X,Y)	3.2.2	YES-2	
3/4.2.4	***Quadrant Power Tilt Ratio***			
3.2.4	Quadrant Power Tilt Ratio	3.2.4	YES-2	
3/4.2.5	***DNB Parameters***			
3.2.5	DNB Parameters	3.4.1	YES-2	
3/4.3	**INSTRUMENTATION**	**3.3**		
3/4.3.1	***Reactor Trip System Instrumentation***			
3.3.1.1 U1 3.3.1 U2	Reactor Trip System Instrumentation	3.3.1 3.3.2 3.3.9	YES-3	
3/4.3.2	***Engineered Safety Feature Actuation System Instrumentation***			
3.3.2.1 U1 3.2.1 U2	Engineered Safety Feature Actuation System Instrumentation	3.3.2 3.3.6	YES-3	

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

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.3.3	***Monitoring Instrumentation***			
3.3.3.1	Radiation Monitoring Instrumentation	3.3.6 3.3.7 3.3.8	YES-3	
Instrument 1	Area Monitors			
Instrument 1.a	Fuel Storage Pool Area Emergency Ventilation System Actuation	3.3.8	YES-3	
Instrument 2	Process Monitors			
Instrument 2.a	Containment Purge Air	3.3.6	YES-3	
Instrument 2.b.ii	Containment Particulate Activity RCS Leakage Detection	3.4.15	YES-1	
Instrument 2.c	Control Room Isolation	3.3.7	YES-3	
3.3.3.5	Remote Shutdown Instrumentation	3.3.4	YES-4	
3.3.3.7	Accident Monitoring Instrumentation	3.3.3 5.6.5	YES-3	
3.3.3.10	Explosive Gas Monitoring Instrumentation	Relocated	NO	See Appendix A, Page 1.
3.3.3.11	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	3.3.5	YES-3	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.4	**REACTOR COOLANT SYSTEM**	**3.4**		
3/4.4.1	***Reactor Coolant Loops and Coolant Circulation***			
3.4.1.1	Startup and Power Operation	3.4.4	YES-2	
3.4.1.2	Hot Standby	3.4.5	YES-3	
3.4.1.3	Shutdown	3.4.6	YES-4	
3.4.1.4	Cold Shutdown	3.4.7 3.4.8	YES-4	
3/4.4.3	***Safety and Relief Valve - Operating***			
3.4.3.1	Safety Valves-Operating	3.4.10	YES-3	
3.4.3.2	Relief Valves -Operating	3.4.11	YES-3	
3/4.4.4	***Pressurizer***			
3.4.4	Pressurizer	3.4.9	YES-2	
3/4.4.5	***Steam Generators***			
3.4.5	Steam Generators	3.4.17 5.5.9	YES-2	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.4.6	***Reactor Coolant System Leakage***			
3.4.6.1	Leakage Detection Instrumentation	3.4.15	YES-1	
3.4.6.2	Operational Leakage	3.4.13	YES-2	
3.4.6.3	Reactor Coolant System Pressure isolation Valve Leakage	3.4.14	YES-2	
3/4.4.8	***Specific Activity***			
3.4.8	Specific Activity	3.4.16	YES-2	
3/4.4.9	***RCS Pressure and Temperature (PT)Limits***			
3.4.9.1	RCS Pressure and Temperature (PT)Limits	3.4.3	YES-2	
3/4.4.12	***Low Temperature Over Pressure Protection (LTOP) System***			
3.4.12	Low Temperature Over Pressure Protection (LTOP) System	3.4.12	YES-2	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.5	**EMERGENCY CORE COOLING SYSTEMS(ECCS)**	3.5		
3/4.5.1	***Accumulators***			
3.5.1	Cold Leg Injection Accumulators	3.5.1	YES-3	
3/4.5.2	***ECCS Subsystems - Operating***			
3.5.2	ECCS Subsystems - Operating	3.5.2	YES-3	
3/4.5.3	***ECCS Subsystems - Shutdown***			
3.5.3	ECCS Subsystems - Shutdown	3.5.3	YES-3	
3/4.5.5	***Refueling Water Storage Tank***			
3.5.5	Refueling Water Storage Tank	3.5.4	YES-3	
3/4.5.6	***Seal Injection Flow***			
3.5.6	Seal Injection Flow	3.5.5	YES-2	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.6	**CONTAINMENT SYSTEMS**	**3.6**		
3/4.6.1	***Primary Containment***			
3.6.1.1	Containment Integrity	3.6.1 3.6.2	YES-3	
3.6.1.3	Containment Air Locks	3.6.2	YES-3	
3.6.1.4	Internal Pressure	3.6.4	YES-2	
3.6.1.5	Air Temperature	3.6.5	YES-2	
3.6.1.6	Containment Vessel Structural Integrity	3.6.1	YES-3	Containment vessel structural integrity is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1.
3.6.1.7	Shield Building Structural Integrity	3.6.7	YES-3	
3.6.1.8	Emergency Gas Treatment System-EGTS-Clean Up Subsystems	3.6.10 5.5.9	YES-3	
3/4.6.2	***Depressurization and Cooling Systems***			
3.6.2.1	Containment Spray SubSystems	3.6.6	YES-3	
3.6.2.2	Lower Containment Vent Coolers	Relocated	No	See Appendix A, Page 2.

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.6.3	***Containment Isolation Valves***	***3.6.3***	***YES-3***	
3.6.3	Containment Isolation Valves	3.6.3	YES-3	
3/4.6.4	***Combustible Gas Control***			
3.6.4.3	Hydrogen Mitigation System	3.6.8	YES-4	
3/4.6.5	***Ice Condenser***			
3.6.5.1	Ice Bed	3.6.12	YES-3	
3.6.5.3	Ice Condenser Doors	3.6.13	YES-3	
3.6.5.5	Divider Barrier Personnel Access Doors and Equipment hatches	3.6.14	YES-3	
3.6.5.6	Containment Air Return Fans	3.6.11	YES-3	
3.6.5.7	Floor Drains	3.6.15	YES-3	
3.6.5.8	Refueling Canal Drains	3.6.15	YES-3	
3.6.5.9	Divider Barrier Seals	3.6.14	YES-3	
3/4.6.6	***Vacuum Relief Lines***			
3.6.6	Vacuum Relief lines	3.6.9	YES-3	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.7	**PLANT SYSTEMS**	**3.7**		
3/4.7.1	***Turbine Cycle***			
3.7.1.1	Safety Valves	3.7.1	YES-3	
3.7.1.2	Auxiliary Feedwater (AFW) System	3.7.5	YES-3	
3.7.1.3	Condensate Storage System	3.7.6	YES-2, 3	
3.7.1.4	Activity	3.7.16	YES-2	
3.7.1.5	Main Steam Line Isolation Valves	3.7.2	YES-3	
3.7.1.6	Main Feedwater Isolation, Regulation and Bypass Valves	3.7.3	YES-3	
3/4.7.3	***Component Cooling Water System***			
3.7.3	Component Cooling Water System	3.7.7	YES-3	
3/4.7.4	***Service Water System***			
3.7.4	Service Water System	3.7.8	YES-3	
3/4.7.5	***Ultimate Heat Sink***			
3.7.5	Ultimate Heat Sink	3.7.9	YES-3	

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.7.7	***Control Room Emergency Ventilation System***			
3.7.7	Control Room Emergency Ventilation System	3.7.10 5.5.9	YES-3	
3/4.7.8	***Auxiliary Building Gas Treatment System***			
3.7.8	Auxiliary Building Gas Treatment System	3.7.12 5.5.9	YES-3	
3/4.7.13	***Spent Fuel Pool Minimum Boron Concentration***			
3.7.13	Spent Fuel Pool Minimum Boron Concentration	3.7.14	YES-2	
3/4.7.14	***Cask Pit Pool Minimum Boron Concentration***			
3.7.14	Cask Pit Pool Minimum Boron Concentration	3.7.17	YES-2	
3/4.7.15	***Control Room Air Conditioning System***			
3.7.15	Control Room Air Conditioning System	3.7.11	YES-3	

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

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.8	**ELECTRICAL POWER SYSTEM**	**3.8**		
3/4.8.1	***A.C. Sources***			
3.8.1.1	Operating	3.8.1 3.8.3 3.8.4 3.8.6 3.8.9	YES-3	
3.8.1.2	Shutdown	3.8.2 3.8.3 3.8.5 3.8.6 3.8.10	YES-3	
3/4.8.2	***Onsite Power Distribution Systems***			
3.8.2.1	A.C. Distribution - Operating	3.8.7 3.8.9	YES-3	
3.8.2.2	A.C. Distribution - Shutdown	3.8.8 3.8.10	YES-3	
3.8.2.3	D.C. Distribution - Operating	3.8.4 3.8.6 3.8.9	YES-3	

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

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.8.2.4	D.C. Distribution - Shutdown	3.8.5 3.8.6 3.8.10	YES-3	
3/4.9	**REFUELING OPERATIONS**	**3.9**		
3/4.9.1	***Boron Concentration***			
3.9.1	Boron Concentration	3.9.1 3.9.2	YES-2	
3/4.9.2	***Instrumentation***			
3.9.2	Instrumentation	3.9.3	YES-3	
3/4.9.3	***Decay Time***			
3/4.9.3	Decay Time	~~Deleted~~ 3.9.8	~~NO~~ YES-2	~~See technical change discussion in Enclosure 2, Volume 14, Discussion of Changes for CTS 3/4.9.3.~~
3/4.9.4	***Containment Building Penetrations***			
3.9.4	Containment Building Penetrations	3.9.4	YES-3	

SII

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.9.8	***Residual Heat Removal and Coolant Circulation***			
3.9.8.1	All Water Levels	3.9.5 3.9.6	YES-4	
3.9.8.2	Low Water Level	3.9.6	YES-4	
3/4.9.9	***Containment Ventilation Isolation System***			
3.9.9	Containment Ventilation Isolation System	3.9.4	YES-3	
3/4.9.10	***Water Level - Reactor Vessel***			
3.9.10	Water Level - Reactor Vessel	3.9.7	YES-2	
3/4.9.11	***Storage Pool Water Level***			
3.9.11	Storage Pool Water Level	3.7.13	YES-2, 3	
3/4.9.12	***Auxiliary Building Gas Treatment System***			
3.9.12	Auxiliary Building Gas Treatment System	3.7.12 5.5.9	YES-3	

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

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.10	**SPECIAL TEST EXCEPTIONS**			
3/4.10.1	***Shutdown Margin***			
3.10.1	Shutdown Margin	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.1.
3/4.10.2	***Group Height, Insertion and Power Distribution Limits***			
3/4.10.2	Group Height, Insertion and Power Distribution Limits	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.2.
3/4.10.3	***Physics Tests***			
3.10.3	Physics Tests	3.1.8	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained to provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs.

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

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.10.4	***Reactor Coolant Loops***			
3.10.4	Reactor Coolant Loops	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.4.
3/4.11	**RADIOACTIVE EFFLUENTS**	**NA**		
3/4.11.1	***Liquid Effluents***			
3.11.1.4	Liquid Holdup Tanks	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
3/4.11.2	***Gaseous Effluents***			
3.11.2.5	Explosive Gas Mixture	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.

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

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.11.2.6	Gas Decay Tanks	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
5.0	**DESIGN FEATURES**	**3.7.15** **4.0**	YES-2 YES	 Application of Technical Specification selection criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
6.0	**ADMINISTRATIVE CONTROLS**	**5.0**	YES	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

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

APPENDIX A

JUSTIFICATION FOR SPECIFICAITON RELOCATION

Appendix A - Justification For Specification Relocation

3.3.3.10, Explosive Gas Monitoring Instrumentation

DISCUSSION:

CTS 3.3.3.10 provides the requirements for the explosive gas monitoring instrumentation. This Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous waste processing system is adequately monitored to ensure that the concentration is maintained below the flammability limit.

COMPARISON TO SCREENING CRITERIA:

1. Explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Explosive gas monitoring instrumentation is not used to indicate the status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. In addition, excessive system oxygen is not an indication of a DBA or transient.

3. Explosive gas monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient. In addition, excessive oxygen discharge is not part of a primary success path in mitigating a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the loss of the explosive gas monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. TVA has reviewed this evaluation, considers it applicable to Sequoyah Nuclear Plant (SQN) Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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Appendix A - Justification For Specification Relocation

3.6.2.2, Lower Containment Vent Coolers

DISCUSSION:

CTS 3.6.2.2 provides requirements on the Lower Containment Vent Coolers. The Lower Containment Vent Coolers are designed to maintain an acceptable temperature within the lower containment compartments for the protection of equipment and controls during normal reactor operation and normal shutdown.

COMPARISON TO SCREENING CRITERIA:

1. The Lower Containment Vent Coolers are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Lower Containment Vent Coolers are not a process variable, design feature, or operating restriction that is in an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Lower Containment Vent Coolers are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The Lower Containment Vent Coolers were found to be non-significant risk contributor to core damage frequency and offsite releases. Tennessee Valley Authority (TVA) has performed a plant-specific analysis to ensure that the Lower Containment Vent Coolers do not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to be important to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Lower Containment Vent Coolers may be relocated to other plant controlled documents outside Technical Specifications.

ENCLOSURE 2

VOLUME 3

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 1.0
USE AND APPLICATION

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 1.0, USE AND APPLICATION

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

ITS

Chapter 1.0

Definitions 1.1

A01

1.0 USE AND APPLICATION

1

1.~~0~~ DEFINITIONS

~~DEFINED TERMS~~

NOTE

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

and Bases

A01

Term | Definition

ACTIONS

ACTION

S

that

Required Actions to be taken

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

within specified Completion Times

INSERT 1

A02

AXIAL FLUX DIFFERENCE (AFD)

AXIAL FLUX DIFFERENCE

(AFD)

AFD

A01

~~1.2~~ ~~AXIAL FLUX DIFFERENCE~~ shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

BYPASS LEAKAGE PATH

1.3 A BYPASS LEAKAGE PATH is a potential path for leakage to escape from both the primary containment and annulus pressure boundary. Only one type of BYPASS LEAKAGE PATH is recognized:

a. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING are those paths that would potentially allow leakage from the primary containment to circumvent the annulus secondary containment enclosure and escape directly to the Auxiliary Building secondary containment enclosure.

See ITS 3.6.3

CHANNEL CALIBRATION

CHANNEL CALIBRATION

that

A01

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

within

INSERT 2

A03

means of

,

A01

CHANNEL CHECK

CHANNEL CHECK

, by observation,

A01

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

to

A02

INSERT 1

ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.

A03

INSERT 2

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

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DEFINITIONS

OPERATIONAL

CHANNEL OPERATIONAL TEST

CHANNEL ~~FUNCTIONAL~~ TEST (COT)

A04

A01

~~1.6~~ A ~~CHANNEL FUNCTIONAL TEST~~ COT shall be~~:~~

L01

or actual

a. ~~Analog channels -~~ the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions~~.

INSERT 3

A04

b. ~~Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.~~

A05

c. ~~Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.~~

A04

~~CONTAINMENT INTEGRITY~~

~~1.7 CONTAINMENT INTEGRITY shall exist when:~~

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

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

~~2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.~~

~~b. All equipment hatches are closed and sealed.~~

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

~~d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,~~

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

~~f. Secondary containment bypass leakage is within the limits of Specification 3.6.3.~~

A06

KAB-066

~~CONTROLLED LEAKAGE~~

~~1.8 This definition has been deleted.~~

A07

stet

CORE ALTERATION

~~CORE ALTERATION~~

~~1.9~~ ~~CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~

A01

~~A06~~

CORE OPERATING LIMITS REPORT

S

CORE OPERATING LIMIT REPORT (COLR)

parameter

~~1.10~~ The ~~CORE OPERATING LIMITS REPORT (~~COLR~~)~~ is the unit~~-~~specific document that provides ~~core operating~~ limits for the current ~~operating~~ reload cycle. These cycle~~-~~specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.14. Unit~~ operation within these ~~operating~~ limits is addressed in individual specifications.

cycle specific parameter

5.6.3. Plant

A01

A04

INSERT 3

of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.

Enclosure 2, Volume 3, Rev. 0, Page 8 of 117

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131

when inhaled as the combined activities of iodine isotopes

that

A01

~~1.11~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/ gram) ~~which~~ alone would produce the same thyroid dose ~~as the quantity and isotopic mixture of~~ I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid dose conversion factors used for this calculation shall be those listed~~ in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." AEC, 1962,

INSERT 4

determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from

A02

~~Ē - AVERAGE DISINTEGRATION ENERGY~~

~~1.12~~ ~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~

A06

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

ESF

~~1.13~~ The ~~ENGINEERED SAFETY FEATURE~~ RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ~~ESF~~ actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays , where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and ~~the~~ methodology for verification have been previously reviewed and approved by the NRC.

A01

A01

A01

~~FREQUENCY NOTATION~~

~~1.14~~ ~~The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.~~

LA02

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.15~~ ~~A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

A06

LEAKAGE

~~IDENTIFIED~~ LEAKAGE

~~1.16~~ ~~IDENTIFIED~~ LEAKAGE shall be:

A01

a. Identified LEAKAGE

(RCP)

A08

~~a.~~ 1. Leakage, such as that from pump seals or valve packing (except Reactor Coolant Pump Seal Water Injection or Leakoff), that is captured and conducted to collection systems or a sump or collecting tank, ~~or~~

TSTF-490

INSERT 4

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

LEAKAGE

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

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

INSERT 5

A01 A02 A07

~~MEMBER(S) OF THE PUBLIC~~

~~1.17 DELETED~~

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

See ITS 5.5

OPERABLE - OPERABILITY

OPERABLE - OPERABILITY

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

A01 A09 A10

MODE

~~OPERATIONAL MODE -~~ MODE

~~1.20~~ A~~n~~ ~~OPERATIONAL~~ MODE ~~(i.e., MODE)~~ shall correspond to any one inclusive combination of core reactivity condition, power level , ~~and~~ average reactor coolant temperature, INSERT 6 specified in Table 1.1-1 with fuel in the reactor vessel.

A01 A11

PHYSICS TESTS

PHYSICS TESTS

~~1.21~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:
a. ~~and 1)~~ described in Chapter 14.0, Initial Tests and Operations, of the FSAR,
b. ~~2)~~ authorized under the provisions of 10 CFR 50.59, or
c. ~~3)~~ otherwise approved by the Nuclear Regulatory Commission.

A01

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A02

INSERT 5

MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

A11

INSERT 6

, and reactor vessel head closure bolt tensioning

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c.

LEAKAGE

PRESSURE BOUNDARY LEAKAGE

~~1.22~~ ~~PRESSURE BOUNDARY~~ LEAKAGE ~~shall be leakage~~ (except primary to secondary leakage) through a non-isolable fault in ~~a Reactor Coolant System~~ an RCS component body, pipe wall , or vessel wall. A01 A08

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

~~1.23~~ The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the ~~LTOP~~ low temperature overpressure protection arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification ~~6.9.1.15~~ 5.6.4. A01

~~PROCESS CONTROL PROGRAM (PCP)~~ A07

~~1.24~~ ~~DELETED~~

~~PURGE - PURGING~~

~~1.25~~ ~~PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~ A06

QUADRANT POWER TILT RATIO (QPTR)

QUADRANT POWER TILT RATIO (QPTR)

~~1.26~~ ~~QUADRANT POWER TILT RATIO~~ QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. A01

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP)

~~1.27~~ ~~RATED THERMAL POWER (~~RTP~~)~~ shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt. A01

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

~~1.28~~ The ~~REACTOR TRIP SYSTEM~~ RTS RESPONSE TIME shall be ~~the~~ that time interval from when the monitored parameter exceeds its ~~(~~RTS~~)~~ trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and ~~the~~ methodology for verification have been previously reviewed and approved by the NRC. A01

~~REPORTABLE EVENT~~ A07

~~1.29~~ ~~DELETED~~

CET003

3.6.7 3.6.10 3.6.7 See ITS ~~3.6.3~~ ~~3.6.13~~ ~~3.6.1~~

SHIELD BUILDING INTEGRITY

1.30 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit. See ITS ~~3.6.3~~ 3.6.7

b. The emergency gas treatment system is OPERABLE. See ITS ~~3.6.13~~ 3.6.10

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE. See ITS ~~3.6.1~~ 3.6.7

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM) A01

~~1.31~~ ~~SHUTDOWN MARGIN~~ SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming : a. all ~~full length~~ rod cluster control assemblies (~~shutdown and control~~ RCCAs) are fully inserted except for the single ~~rod cluster assembly~~ RCCA of highest reactivity worth, which is assumed to be fully withdrawn. INSERT 7 A12

~~SITE BOUNDARY~~

~~1.32~~ ~~The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.~~ A06

~~SOLIDIFICATION~~

~~1.33~~ ~~Deleted~~ A07

~~SOURCE CHECK~~

~~1.34~~ ~~Deleted~~ A07

INSERT 8 A02

STAGGERED TEST BASIS

STAGGERED TEST BASIS

~~1.35~~ A STAGGERED TEST BASIS shall consist of~~:~~ INSERT 9

~~a.~~ ~~A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,~~ A13

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

THERMAL POWER

THERMAL POWER

~~1.36~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. A01

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A12

INSERT 7

With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

A02

INSERT 8

SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

A13

INSERT 9

the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

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b.

LEAKAGE

UNIDENTIFIED LEAKAGE

All RCP

1.37 ~~UNIDENTIFIED~~ LEAKAGE ~~shall be all leakage~~ (except ~~reactor coolant pump~~ seal water injection or leakoff) that is not IDENTIFIED LEAKAGE~~.~~ , and A01 A08

~~UNRESTRICTED AREA~~

~~1.38~~ ~~An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commerical, institutional, and/or recreational purposes.~~ A06

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.39~~ ~~A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~ A06

~~VENTING~~

~~1.40~~ ~~VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~ A06

INSERT 10 A02

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A02

INSERT 10

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

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ITS

Chapter 1.0

TABLE 1.1-1

TABLE 1.1 -1 (page 1 of 1)

~~OPERATIONAL~~ MODES

A01

A01

MODE (TITLE)	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER~~*~~ (a)	(REACTOR) AVERAGE COOLANT TEMPERATURE (°F)
1. POWER OPERATION	≥0.99	>5~~%~~	~~≥350°F~~ NA
2. STARTUP	≥0.99	≤5~~%~~	~~≥350°F~~ NA
3. HOT STANDBY	<0.99	~~0~~ NA	≥350~~°F~~
4. HOT SHUTDOWN (b)	<0.99	~~0~~ NA	350~~°F~~ > T_{avg} >200~~°F~~
5. COLD SHUTDOWN (b)	<0.99	~~0~~ NA	≤200~~°F~~
6. REFUELING~~**~~ (c)	~~≤0.95~~ NA	~~0~~ NA	~~≤140°F~~ NA

A01

A14

LA01

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

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

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

A01

A11

~~TABLE 1.2~~

~~FREQUENCY NOTATION~~

~~NOTATION~~	~~FREQUENCY~~
~~S~~	~~At least once per 12 hours.~~
~~D~~	~~At least once per 24 hours.~~
~~W~~	~~At least once per 7 days.~~
~~M~~	~~At least once per 31 days.~~
~~Q~~	~~At least once per 92 days~~.
~~SA~~	~~At least once per 184 days.~~
~~R~~	~~At least once per 18 months.~~
~~S/U~~	~~Prior to each reactor startup.~~
~~P~~	~~Completed prior to each release.~~
~~N.A.~~	~~Not applicable.~~

LA02

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

A15

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} Greater Than 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

SHUTDOWN MARGIN (SDM)

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

A12

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

See ITS 3.1.6

*See Special Test Exception 3.10.1

See ITS 3.1.1

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:

SHUTDOWN MARGIN (SDM)

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

A12

b. At least once per 24 hours by consideration of the following factors:

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

See ITS 3.1.1

ITS

Chapter 1.0

Definitions 1.1

A01

1.0 USE AND APPLICATION

1

1.~~0~~ DEFINITIONS

~~DEFINED TERMS~~

NOTE

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

and Bases

A01

Term | Definition

ACTIONS

ACTION S

A01

that

Required Actions to be taken

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

within specified Completion Times

A01

INSERT 1

A02

AXIAL FLUX DIFFERENCE (AFD)

AXIAL FLUX DIFFERENCE (AFD)

AFD

A01

~~1.2~~ ~~AXIAL FLUX DIFFERENCE~~ shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector~~s~~.

BYPASS LEAKAGE PATH

1.3 A BYPASS LEAKAGE PATH is a potential path for leakage to escape from both the primary containment and annulus pressure boundary. Only one type of BYPASS LEAKAGE PATH is recognized:

a. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING are those paths that would potentially allow leakage from the primary containment to circumvent the annulus secondary containment enclosure and escape directly to the auxiliary building secondary containment enclosure.

See ITS 3.6.3

CHANNEL CALIBRATION

CHANNEL CALIBRATION

that

A01

within

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

INSERT 2

A03

means of

,

A01

CHANNEL CHECK

CHANNEL CHECK

, by observation,

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

to

A01

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A02

INSERT 1

ACTUATION LOGIC TEST — An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.

A03

INSERT 2

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

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DEFINITIONS

A04

CHANNEL OPERATIONAL TEST

OPERATIONAL

CHANNEL ~~FUNCTIONAL~~ TEST (COT) A01

COT

1.6 A ~~CHANNEL FUNCTIONAL TEST~~ shall be: L01

or actual

~~a.~~ ~~Analog channels -~~ the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions~~. A04

INSERT 3

~~b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.~~ A05

~~c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.~~ A04

~~CONTAINMENT INTEGRITY~~

~~1.7 CONTAINMENT INTEGRITY shall exist when:~~

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

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

~~2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.~~

~~b. All equipment hatches are closed and sealed.~~

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

~~d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,~~

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

~~f. Secondary containment bypass leakage is within the limits of Specification 3.6.3.~~

A06

~~CONTROLLED LEAKAGE~~

KAB-066

~~1.8 This definition has been deleted.~~ A07

stet

CORE ALTERATION

~~CORE ALTERATION~~

~~1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~ A01 ~~A06~~

CORE OPERATING LIMITS REPORT

CORE OPERATING LIMITS REPORT (COLR)

parameter

~~1.10~~ The ~~CORE OPERATING LIMITS REPORT~~ ~~(~~COLR~~)~~ is the unit~~-~~specific document that provides ~~core operating~~ limits for the current ~~operating~~ reload cycle. These cycle~~-~~specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.14.~~ ~~Unit~~ operation within these ~~operating~~ limits is addressed in individual specifications. A01

cycle specific parameter

5.6.3. Plant

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A04

INSERT 3

of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.

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DEFINITIONS

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131

when inhaled as the combined activities of iodine isotopes

that

~~1.11~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) ~~which~~ alone would produce the same thyroid dose ~~as the quantity and isotopic mixture of~~ I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid dose conversion factors used for this calculation shall be those listed in~~ Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." A01

AEC, 1962,

INSERT 4

determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from A02

~~Ē - AVERAGE DISINTEGRATION ENERGY~~

~~1.12 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~ A06

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

ESF

~~1.13~~ The ~~ENGINEERED SAFETY FEATURE~~ RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ~~ESF~~ actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays , where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and ~~the~~ methodology for verification have been previously reviewed and approved by the NRC. A01

A01

~~FREQUENCY NOTATION~~

~~1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.~~ LA02

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.15 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~ A06

TSTF-490

INSERT 4

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

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DEFINITIONS

LEAKAGE

~~IDENTIFIED~~ LEAKAGE

1.16 ~~IDENTIFIED~~ LEAKAGE shall be: a. Identified LEAKAGE (RCP) A01 A08

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

2. ~~b.~~ Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or A01

3. ~~c.~~ Reactor coolant system (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage). A02

INSERT 5

~~MEMBER(S) OF THE PUBLIC~~

~~1.17 DELETED~~ A07

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

See ITS 5.5

OPERABLE - OPERABILITY

OPERABLE - OPERABILITY

~~1.19~~ A system, subsystem, train, ~~or~~ component , or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s)~~,~~ and when all necessary attendant instrumentation, controls, ~~a~~ normal ~~and an~~ or emergency electrical power ~~source~~, cooling ~~or~~ and seal water, lubrication, and ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). A01 A09 A10

A02

INSERT 5

MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

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DEFINITIONS

MODE

~~OPERATIONAL MODE -~~ MODE

~~1.20~~ A~~n~~ ~~OPERATIONAL~~ MODE ~~(i.e., MODE)~~ shall correspond to any one inclusive combination of core reactivity condition, power level ~~and~~ average reactor coolant temperature specified in Table 1.1.

, INSERT 6 -1 with fuel in the reactor vessel

A01 A11

PHYSICS TESTS

PHYSICS TESTS

, Initial Tests and Operations,

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

. These tests are: a. b. c. Nuclear Regulatory

A01

LEAKAGE

PRESSURE BOUNDARY LEAKAGE

c.

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

an RCS ,

A01 A08

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

low temperature overpressure protection

~~1.23~~ The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the ~~LTOP~~ arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification ~~6.9.1.15~~.

5.6.4

A01

~~PROCESS CONTROL PROGRAM (PCP)~~

~~1.24 DELETED~~

A07

~~PURGE - PURGING~~

~~1.25 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

A06

QUADRANT POWER TILT RATIO (QPTR)

QUADRANT POWER TILT RATIO (QPTR)

QPTR

~~1.26~~ ~~QUADRANT POWER TILT RATIO~~ shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, which~~-~~ever is greater.

A01

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A11 **INSERT 6**

, and reactor vessel head closure bolt tensioning

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DEFINITIONS

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP)

~~1.27 RATED THERMAL POWER~~ ~~(~~RTP~~)~~ shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt. A01

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

RTS that

~~1.28~~ The ~~REACTOR TRIP SYSTEM~~ RESPONSE TIME shall be ~~the~~ time interval from when the monitored parameter exceeds its ~~(~~RTS~~)~~ trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and ~~the~~ methodology for verification have been previously reviewed and approved by NRC. A01

the

~~REPORTABLE EVENT~~

~~1.29 DELETED~~ CET003 A07

SHIELD BUILDING INTEGRITY

1.30 SHIELD BUILDING INTEGRITY shall exist when:

3.6.7 3.6.10 3.6.7 See ITS ~~3.6.3~~ ~~3.6.13~~ ~~3.6.1~~

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit. See ITS ~~3.6.3~~ 3.6.7

b. The emergency gas treatment system is OPERABLE. See ITS ~~3.6.13~~ 3.6.10

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE. See ITS ~~3.6.1~~ 3.6.7

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM)

SDM control

: a.

~~1.31 SHUTDOWN MARGIN~~ shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full length~~ rod cluster assemblies (~~shutdown and control~~) are fully inserted except for the single ~~rod cluster assembly~~ of highest reactivity worth which is assumed to be fully withdrawn. A01 A12

RCCAs , INSERT 7 RCCA

~~SITE BOUNDARY~~

~~1.32 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.~~ A06

A12

INSERT 7

With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

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DEFINITIONS

~~SOLIDIFICATION~~

~~1.33 Deleted.~~ A07

A07

~~SOURCE CHECK~~

~~1.34 Deleted.~~

STAGGERED TEST BASIS

INSERT 8 A02

STAGGERED TEST BASIS

~~1.35~~ A STAGGERED TEST BASIS shall consist of: INSERT 9

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

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

A13

THERMAL POWER

THERMAL POWER

~~1.36~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. A01

LEAKAGE

b.

UNIDENTIFIED LEAKAGE

All

RCP

~~1.37~~ ~~UNIDENTIFIED~~ LEAKAGE ~~shall be all leakage~~ (except ~~reactor coolant pump~~ seal water injection or leakoff) that is not IDENTIFIED LEAKAGE~~.~~ , and A01 A08

~~UNRESTRICTED AREA~~

~~1.38 An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional, and/or recreational purposes.~~ A06

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A02

INSERT 8

SLAVE RELAY TEST

A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

A13

INSERT 9

the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

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DEFINITIONS

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.39 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~ A06

~~VENTING~~

~~1.40 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~ A06

INSERT 10 A02

A02

INSERT 10

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

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TABLE 1.1-1

TABLE 1.1 -1 (page 1 of 1)

~~OPERATIONAL~~ MODES

TITLE (a) REACTOR (^{0}F)

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER~~*~~ (a)	REACTOR AVERAGE COOLANT TEMPERATURE (^{0}F)
1. POWER OPERATION	≥ 0.99	> 5~~%~~	~~≥ 350°F~~ NA
2. STARTUP	≥ 0.99	≤ 5~~%~~	~~≥ 350°F~~ NA
3. HOT STANDBY (b)	< 0.99	~~0~~ NA	≥ 350~~°F~~
4. HOT SHUTDOWN (b)	< 0.99	~~0~~ NA	350~~°F~~ > T_{avg} > 200~~°F~~
5. COLD SHUTDOWN (b)	< 0.99	0 NA	≤ 200~~°F~~
6. REFUELING~~**~~ (c)	~~≤ 0.95~~ NA	0 NA	~~≤ 140°F~~ NA

A01 A01 A01 A14 LA01

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

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

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

A01 A11

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~~TABLE 1.2~~

~~FREQUENCY NOTATION~~

~~NOTATION~~	~~FREQUENCY~~
~~S~~	~~At least once per 12 hours.~~
~~D~~	~~At least once per 24 hours.~~
~~W~~	~~At least once per 7 days.~~
~~M~~	~~At least once per 31 days~~
~~Q~~	~~At least once per 92 days.~~
~~SA~~	~~At least once per 184 days.~~
~~R~~	~~At least once per 18 months.~~
~~S/U~~	~~Prior to each reactor startup.~~
~~P~~	~~Completed prior to each release.~~
~~N.A.~~	~~Not applicable.~~

LA02

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

A15

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \geq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

A12

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2, with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

See ITS 3.1.6

* See Special Test Exception 3.10.1

See ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

A12

b. At least once per 24 hours by consideration of the following factors:

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

See ITS 3.1.1

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DISCUSSION OF CHANGES
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ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN), Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional approved Technical Specification Task Force (TSTF) travelers included in the submittal.

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

A02 ITS Section 1.1 provides definitions of ACTUATION LOGIC TEST, MASTER RELAY TEST, SLAVE RELAY TEST, DOSE EQUIVALENT XENON XE-133, and TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT). These terms are used as defined terms in the ITS but do not appear in the CTS. This changes the CTS by adding new definitions

This change is acceptable because these new defined terms, of themselves, do not impose any new requirements or alter existing requirements. Any technical changes due to the addition of these defined terms are addressed in the discussion of changes (DOCs) for the sections of the Technical Specifications in which the terms are used. These changes are designated as administrative as they add defined terms that do not involve a technical change to the Technical Specifications.

A03 CTS 1.4 defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated." ITS defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps." This results in a number of changes to the CTS.

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

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

- The CTS states that the CHANNEL CALIBRATION "shall include the CHANNEL FUNCTIONAL TEST." The ITS does not include this statement.

  This change is acceptable because the eliminated CTS statement does not add any requirements. In both the CTS and the ITS, performance of a single test that fully meets the requirements of another test can be credited as satisfying that other test.

- The ITS adds the statement, "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel." The purpose of a CHANNEL CALIBRATION is to adjust the channel output so that the channel responds within the necessary range and accuracy to known values of the parameters that the channel monitors.

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

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

A04 CTS Section 1.0 defines CHANNEL FUNCTIONAL TEST as: "a. Analog channels – the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions; b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions; c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions ." ITS Section 1.1 renames and combines the CTS definition to CHANNEL OPERATIONAL TEST (COT), and defines it as "the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the

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required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps." This changes the CTS by stating that the COT shall include adjustments, as necessary, of the devices in the channel so that the setpoints are within the required range and accuracy, changes the CTS by combining the type of devices contained in the definition, and states that the test may be performed by means of any series of sequential, overlapping, or total channel steps. The addition of use of an actual signal is discussed in DOC L01.

- The CTS definition states that the CHANNEL FUNCTIONAL TEST shall verify that the channel is OPERABLE "including alarm and/or trip functions." Similarly, the ITS requirement states that the COT shall verify OPERABILITY of "all devices in the channel required for channel OPERABILITY."

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

- The ITS requirement states "The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy."

  This change is acceptable because it clarifies that adjustments performed during a COT do not invalidate the test. This is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST and does not result in a technical change to the Technical Specifications.

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

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

- CTS Section 1.0 defines CHANNEL FUNCTIONAL TEST for analog channels and digital channels. The ITS definition combines theses definitions.

  This change is acceptable because it states current Industry practice and is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST. This conclusion was confirmed when the NRC issued SQN Unit 1/Unit 2 License Amendment 140/132 (ADAMS Accession Nos. ML013310103 / ML013330076) concluding that the addition of the definition

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to the CHANNEL FUNCTIONAL TEST for digital channels was consistent with the existing channel functional test definition and therefore acceptable.

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

A05 CTS Section 1.0 includes a CHANNEL FUNCTIONAL TEST definition for bistable channels. The definition of CHANNEL FUNCTIONAL TEST for bistable channels requires "the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions." However, this CTS definition is essentially duplicative of the TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) definition. ITS Section 1.1 does not include this definition, since the requirements for bistable channels are covered by the TADOT definition.

This change is acceptable because the TADOT definition adequately covers bistable channels, and does not impose any new requirements or alter any existing requirements. This change is categorized as administrative because the bistable portion of the definition is duplicative of the TADOT definition.

A06 CTS Section 1.0 includes the following definitions:

- CONTAINMENT INTEGRITY
- GASEOUS RADWASTE TREATMENT SYSTEM
- PURGE – PURGING
- SITE BOUNDARY
- UNRESTRICTED AREA
- VENTILATION EXHAUST TREATMENT SYSTEM
- VENTING
- $\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

KAB-066
- ~~CORE ALTERATION~~

The ITS does not use this terminology and ITS Section 1.1 does not contain these definitions.

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

A07 CTS Section 1.0 shows the following definitions as being deleted:

- CONTROLLED LEAKAGE
- MEMBER(S) OF THE PUBLIC
- PROCESS CONTROL PROGRAM (PCP)
- REPORTABLE EVENT
- SOLIDIFICATION
- SOURCE CHECK

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The ITS does not use this terminology and ITS Section 1.1 does not contain these definitions.

These changes are acceptable because the terms are not used as defined terms in the ITS. Previous license amendments have deleted these definitions. This change removes the placeholder showing these definitions as deleted. These changes are designated as administrative because they eliminate deleted defined terms that are no longer used.

A08 CTS Section 1.0 provides definitions for IDENTIFIED LEAKAGE, PRESSURE BOUNDARY LEAKAGE, and UNIDENTIFIED LEAKAGE. ITS Section 1.1 includes these requirements in one definition called LEAKAGE (which includes three categories: identified LEAKAGE, unidentified LEAKAGE, and pressure boundary LEAKAGE). This changes the CTS by incorporating the definitions into the ITS LEAKAGE definition with no technical changes.

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

A09 The CTS Section 1.0 definition of OPERABLE - OPERABILITY requires a system, subsystem, train, component, or device to be capable of performing its "specified function(s)" and all necessary support systems to also be capable of performing their "function(s)." The ITS Section 1.1 definition of OPERABLE - OPERABILITY requires the system, subsystem, train, component, or device to be capable of performing the "specified safety function(s)," and requires all necessary support systems that are required for the system, subsystem, train, component, or device to perform its "specified safety function(s)" to also be capable of performing their related support functions. This changes the CTS by altering the requirement to be able to perform "functions" to a requirement to be able to perform "safety functions."

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

A10 The CTS Section 1.0 definition of OPERABLE - OPERABILITY requires that all necessary normal and emergency electrical power sources be available for the system, subsystem, train, component, or device to be OPERABLE. The ITS Section 1.1 definition of OPERABLE - OPERABILITY will replace the phrase "normal and emergency electrical power sources" with "normal or emergency electrical power." This changes the CTS definition of OPERABLE - OPERABILITY by allowing a device to be considered OPERABLE with either normal or emergency power available.

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The OPERABILITY requirements for normal and emergency power sources are addressed in CTS 3.0.5. These requirements allow only the normal or the emergency electrical power source to be OPERABLE, provided its redundant system(s), subsystem(s), train(s), component(s), and device(s) (redundant to the systems, subsystems, trains, components, and devices with an inoperable power source) are OPERABLE. This effectively changes the current "and" to an "or." The existing CTS 3.0.5 requirements are incorporated into ITS 3.8.1 ACTIONS for when a normal (offsite) or emergency (diesel generator) power source is inoperable. Therefore, the ITS definition now uses the word "or" instead of the current word "and." In ITS 3.8.1, new times are provided to perform the determination of OPERABILITY of the redundant systems. This change is discussed in the Discussion of Changes (DOCs) for ITS 3.8.1. This change is designated administrative since the ITS definition is effectively the same as the CTS definition.

A11 CTS Section 1.0 and Table 1.1, "OPERATIONAL MODES," provide a description of the MODES. CTS Section 1.0 and Table 1.1 contains Note ** that states, "Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed." ITS Section 1.1 and Table 1.1-1, "MODES," changes the CTS MODE definitions in the following ways:

- The CTS Table 1.1 Note ** condition "fuel in the vessel" is moved to the ITS MODE definition.

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

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

  This change is acceptable because the revised phrase is consistent with the current interpretation and usage. MODE 6 is currently declared when the first vessel head closure bolt is detensioned. This change also eliminates a redundant phrase. The reactor vessel head cannot be removed unless the reactor vessel head closure bolts are unbolted and they cannot be unbolted unless they are detensioned. Since “reactor vessel head unbolted” is already specified in the CTS Note, including “or removed” is unnecessary.

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

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

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

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

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

A12 CTS Section 1.0 provides a definition of SHUTDOWN MARGIN (SDM). The ITS Section 1.1 definition of SDM contains two differences from the CTS definition.

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

  This change is acceptable because it is consistent with the existing SDM requirements in CTS 3.1.1.1 and 3.1.1.2.

- The CTS definition is clarified to include a description of the reactor fuel and moderator temperature conditions (i.e., nominal zero power level) at which the SDM is calculated when in MODE 1 or 2.

  This change is acceptable because including this information is not a technical change. SDM calculations are currently performed for nominal zero power conditions.

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

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A13 The CTS Section 1.0 definition of STAGGERED TEST BASIS states, "A STAGGERED TEST BASIS shall consist of: a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval." The ITS Section 1.1 definition states, "A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function." This changes the CTS to specify the frequency of a Surveillance on one system, subsystem, train, or other designated component in the Frequency column of the ITS instead of specifying the frequency in which all systems, subsystems, trains, or other designated components must be tested.

This change is acceptable because the testing frequency of components on a STAGGERED TEST BASIS is not changed. Unlike the CTS definition, the ITS definition allows the Surveillance interval for one subsystem to be specified in the Frequency column of the applicable Surveillance Requirements, independent of the number of subsystems. As an example, consider a three-channel system tested on a STAGGERED TEST BASIS. The CTS would specify testing every three months on a STAGGERED TEST BASIS, which results in one channel being tested each month (three equal subintervals). Under the ITS definition, the Surveillance Frequency would be monthly on a STAGGERED TEST BASIS and, one channel would be tested each month. In both the CTS and ITS definitions, all channels are tested every three months. Each test under the CTS definition would be performed at the beginning of the subinterval. Under the ITS definition, each Surveillance Frequency starts at the beginning of the CTS definition subinterval. Thus, there are no net changes in the testing interval. This change represents an editorial preference in the ITS. This change is designated as administrative as no technical changes are made to the Technical Specifications.

A14 CTS Table 1.1, OPERATIONAL MODES, is revised. The corresponding table in ITS Section 1.1 is Table 1.1-1, MODES. The changes to the CTS are:

- The CTS Table 1.1 minimum average reactor coolant temperature for MODES 1 and 2 is changed from ≥ 350°F to "NA" (not applicable) in ITS Table 1.1-1.

  This change is acceptable because ITS LCO 3.4.2, RCS Minimum Temperature for Criticality, provides the minimum reactor coolant temperature limits for MODES 1 and 2. Therefore, the 350°F minimum temperature does not provide any useful information in ITS Table 1.1-1, and is deleted from the CTS.

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

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

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

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

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

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

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

A15 ITS Sections 1.2, 1.3, and 1.4 contain information that is not in the CTS. This change to the CTS adds explanatory information on ITS usage that is not applicable to the CTS. The added sections are:

- <u>Section 1.2 - Logical Connectors</u>

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

- <u>Section 1.3 - Completion Times</u>

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

- <u>Section 1.4 – Frequency</u>

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

This change is acceptable because it aids in the understanding and use of the format and presentation style of the ITS. The addition of these sections does not add or delete technical requirements, and will be discussed specifically in those

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Technical Specifications where application of the added sections results in a change. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 1.1, "OPERATIONAL MODES," states that MODE 6 is restricted to reactivity conditions with $k_{eff} \leq 0.95$. ITS Table 1.1-1, "MODES," does not contain this restriction.

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

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program).* CTS 1.14 and CTS Table 1.2 present Frequency Notation for the performance of Surveillance Requirements in the CTS. The ITS specify the periodic Frequency as "In accordance with the Frequency Control Program." This changes the CTS by moving the Frequency Notation Table to the Surveillance Frequency Control Program.

The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure

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that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to ensure the associated Limiting Conditions for Operations are met. This change is designated as a less restrictive removal of detail change because the Surveillance Frequencies are being removed from the Technical Specifications and placed in a license control document.

LESS RESTRICTIVE CHANGES

L01 The CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 renames the CTS definition to CHANNEL OPERATIONAL TEST (COT) (discussed in DOC A04) and allows the use of a simulated or actual signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance based on the collection of sufficient information to satisfy the surveillance test requirements.

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

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

CTS

CTS 1.0 USE AND APPLICATION

1.0 1.1 Definitions

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

CTS	Term	Definition
1.1	ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
	ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
1.2	AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the ~~[~~top and bottom halves of a two section excore neutron detector~~]~~. 2
1.4	CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
1.5	CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

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CTS

1.1 Definitions

1.6	CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
	KAB-066	
1.9	CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
1.10	CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
1.11	DOSE EQUIVALENT I-131	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].~~ INSERT 1
	~~Ē - AVERAGE DISINTEGRATION ENERGY~~	~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.~~

7

TSTF-490

TSTF-490

TSTF-490

TSTF-490

INSERT 1

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using

~~------------------------ Reviewer's Note ------------------------------~~
~~The thyroid dose conversion factors to be listed are those assumed in the steam generator tube rupture analysis and, if limiting, the steam line break analysis and must be those factors used to calculate the limit in LCO 3.4.16, "RCS Specific Activity." The first set of thyroid dose conversion factors shall be used for plants licensed to 10 CFR 100.11. The following Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) conversion factors should be used for plants licensed to 10 CFR 50.67.~~
~~--~~

~~[~~thyroid dose conversion factors from~~:~~

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

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

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

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

~~OR~~

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

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INSERT 1 (continued)

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

RPG-013

2

CTS

1.1 Definitions

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

1.16 a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank~~,~~ ; 4

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE~~,~~ ; or 4

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

1.37 b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE~~,~~ ; and 4

1.22 c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

CTS

1.1 Definitions

DOC A02	MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
1.20	MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
1.19	OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
1.21	PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter ~~[~~14 .0, Initial Test s and Operations, ~~Program]~~ of the U FSAR ~~,~~ ; (2) (4) b. Authorized under the provisions of 10 CFR 50.59 ~~,~~ ; or (4) c. Otherwise approved by the Nuclear Regulatory Commission.
1.23	PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the low temperature overpressure protection arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.

CTS

1.1 Definitions

1.26	QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
1.27	RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2893]~~ 3455 MWt. 2
1.28	REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
1.31	SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. ~~However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation.~~ With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM~~,~~ ; and 6 4 b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the ~~[~~nominal zero power design level~~]~~. 2
DOC A02	SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

CTS

1.1 Definitions

1.35	[STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. ~~]~~ 2 2
1.36	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
DOC A02	TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

CTS

Table 1.1

Table 1.1-1 (page 1 of 1)
MODES

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

2
2
2

(a) Excluding decay heat.

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

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

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1.0 USE AND APPLICATION

1.2 Logical Connectors

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

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

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

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

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

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

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

a. Must exist concurrent with the <u>first</u> inoperability and

1.3 Completion Times

DESCRIPTION (continued)

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

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

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

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

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

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

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

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

EXAMPLE 1.3-2

ACTIONS

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

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

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1.3 Completion Times

EXAMPLES (continued)

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

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

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

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

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

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

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

1.3 Completion Times

EXAMPLES (continued)

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

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

<u>EXAMPLE 1.3-4</u>

ACTIONS

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

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

1.3 Completion Times

EXAMPLES (continued)

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

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

EXAMPLE 1.3-5

ACTIONS

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

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

5

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

1.3 Completion Times

EXAMPLES (continued)

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

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

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

EXAMPLE 1.3-6

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

<u>EXAMPLE 1.3-7</u>

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, The Required Action should be pursued without delay and in a controlled manner. 5

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0~~.2~~, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. 3

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

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

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

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

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

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------------------NOTE--------------------------- Not required to be met in MODE 3. -- Verify parameter is within limits.	24 hours

Example 1.4-~~[~~6~~]~~ specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency ~~were~~ was not met), SR 3.0.4 would require satisfying the SR. 2 5

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CTS

CTS 1.0 USE AND APPLICATION

1.0 1.1 Definitions

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

	Term	Definition
1.1	ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
	ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
1.2	AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the ~~[~~top and bottom halves of a two section excore neutron detector~~]~~. 2
1.4	CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
1.5	CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CTS

1.1 Definitions

1.6	CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
	KAB-066	
1.9	CORE ALTERATION	7
1.10	CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
1.11	DOSE EQUIVALENT I-131	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].~~ INSERT 1 TSTF-490
	~~Ē - AVERAGE DISINTEGRATION ENERGY~~	~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.~~ TSTF-490 TSTF-490

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

TSTF-490

INSERT 1

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using

~~------------------------- Reviewer's Note -----------------------------~~
~~The thyroid dose conversion factors to be listed are those assumed in the steam generator tube rupture analysis and, if limiting, the steam line break analysis and must be those factors used to calculate the limit in LCO 3.4.16, "RCS Specific Activity." The first set of thyroid dose conversion factors shall be used for plants licensed to 10 CFR 100.11. The following Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) conversion factors should be used for plants licensed to 10 CFR 50.67.~~
~~---~~

~~[~~thyroid dose conversion factors from~~:~~

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

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

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

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

~~OR~~

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

TSTF-490

INSERT 1 (continued)

DOSE EQUIVALENT XE-133

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

RPG-013

2

CTS

1.1 Definitions

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

1.16

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank~~,~~ ; 4

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE~~,~~ ; or 4

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

1.37

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE~~,~~ ; and 4

1.22

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

1.1 Definitions

DOC A02 MASTER RELAY TEST — A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

1.20 MODE — A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

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

1.21 PHYSICS TESTS — PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter ~~[~~14 .0, Initial Test s and Operations, ~~Program]~~ of the U FSAR ~~,~~ ; (2) (4)

b. Authorized under the provisions of 10 CFR 50.59 ~~,~~ ; or (4)

c. Otherwise approved by the Nuclear Regulatory Commission.

1.23 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) — The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the low temperature overpressure protection arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.

CTS

1.1 Definitions

1.26	QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
1.27	RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2893]~~ 3455 MWt. 2
1.28	REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
1.31	SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. ~~However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation.~~ With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM~~,~~ ; and 6 4 b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the ~~[~~nominal zero power design level~~]~~. 2
DOC A02	SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

CTS

1.1 Definitions

1.35	[STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. ~~]~~	2 2
1.36	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	
DOC A02	TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.	

CTS

Table 1.1

Table 1.1-1 (page 1 of 1)
MODES

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

(a) Excluding decay heat.

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

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

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

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

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

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

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

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

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

a. Must exist concurrent with the first inoperability and

1.3 Completion Times

DESCRIPTION (continued)

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

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

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

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

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

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

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

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

EXAMPLE 1.3-2

ACTIONS

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

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

1.3 Completion Times

EXAMPLES (continued)

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

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

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

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

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

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

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

1.3 Completion Times

EXAMPLES (continued)

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

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

EXAMPLE 1.3-4

ACTIONS

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

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

1.3 Completion Times

EXAMPLES (continued)

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

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

EXAMPLE 1.3-5

ACTIONS

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

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

5

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

1.3 Completion Times

EXAMPLES (continued)

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

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

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

EXAMPLE 1.3-6

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

EXAMPLE 1.3-7

ACTIONS

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

1.3 Completion Times

EXAMPLES (continued)

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

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

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, ~~T~~the Required Action should be pursued without delay and in a controlled manner. 5

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0~~.2~~, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. 3

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

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

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

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

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

EXAMPLES

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

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

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

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

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

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

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

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE--------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

Example 1.4-~~[~~6~~]~~ specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency ~~were~~ was not met), SR 3.0.4 would require satisfying the SR. 2 5

JUSTIFICATION FOR DEVIATIONS
ITS 1.0, USE AND APPLICATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Typographical error is corrected. The proper section for Surveillance Requirement (SR) Applicability is Section 3.0.

4. These punctuation corrections have been made consistent with the Writers Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

5. Typographical error is corrected.

6. The ISTS definition of Shutdown Margin states in part, "However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation." The CTS definition of Shutdown Margin does not contain this allowance, therefore the ITS does not include this allowance. This is acceptable since the information is changed to reflect the current licensing basis.

KAB-066

7. The ISTS does not contain a definition for CORE ALTERATION. The CTS definition for CORE ALTERATION has been included in ITS. This change is acceptable because the information reflects the current licensing basis.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L01

The Tennessee Valley Authority (TVA) is converting Sequoyah to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, Rev. 4, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of no significant hazards considerations for conversion to NUREG-1431.

The CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 renames the CTS definition to CHANNEL OPERATIONAL TEST (COT) and allows the use of a simulated or actual signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance based on the collection of sufficient information to satisfy the surveillance test requirements.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal, and the proposed requirement does not change the technical content or validity of the test. This change will not affect the probability of an accident. The source of the signal sent to components during a Surveillance is not assumed to be an initiator of any analyzed event. The consequence of an accident is not affected by this change. The results of the testing, and, therefore, the likelihood of discovering an inoperable component, are unaffected. As a result, the assurance that equipment will be available to mitigate the consequences of an accident is unaffected.

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

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

   Response: No.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 1.0, USE AND APPLICATION

The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. This change will not physically alter the plant (no new or different type of equipment will be installed). The change does not require any new or revised operator actions.

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

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

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. The margin of safety is not affected by this change. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. As a result, the proposed requirement does not change the technical content or validity of the test.

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

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

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ENCLOSURE 2

VOLUME 4

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Chapter 2.0, SAFETY LIMITS (SLs)

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

A01

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 **2.1 SAFETY LIMITS**

REACTOR CORE

STET SII

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (Tavg) shall not exceed the limits ~~shown in Figure 2.1-1~~ ~~in the COLR~~ and the following SLs shall not be exceeded:. ~~LA01~~

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

2.1.1.2 The maximum local fuel pin centerline temperature shall be maintained ≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications.

Applicability APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 If SL 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

Applicability APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

2.2.2.1 MODES 1 and 2

2.2.2.1 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

2.2.2.2 MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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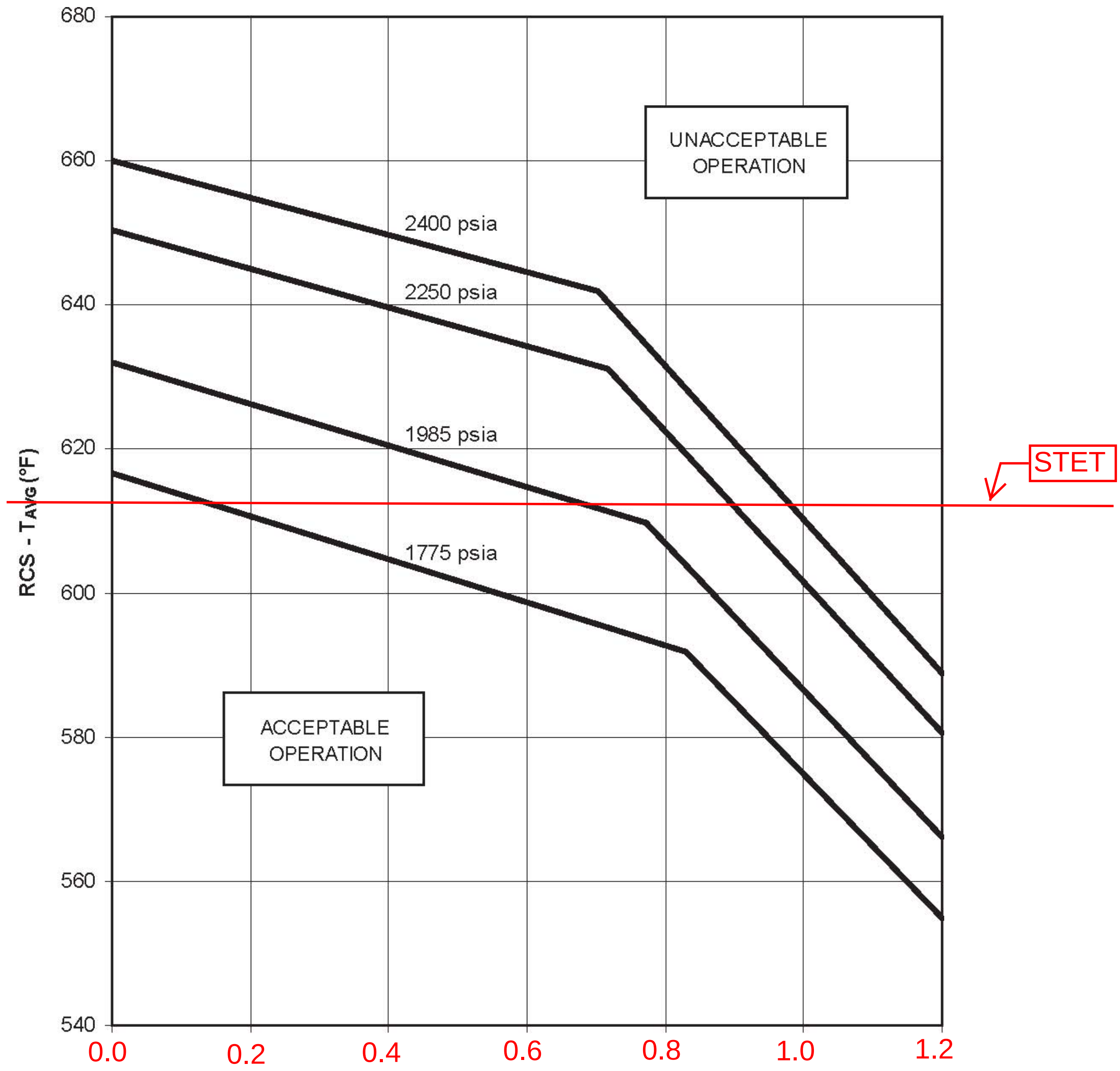
A01

STET

Figure 2.1.1-1 ~~Figure 2.1-1 Reactor Core Safety Limit - Four Loops in Operation~~



FRACTION OF RATED THERMAL POWER

SII

~~LA01~~

SII

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 2.1 SAFETY LIMITS

REACTOR CORE

SII

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Applicability APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 If SL 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

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2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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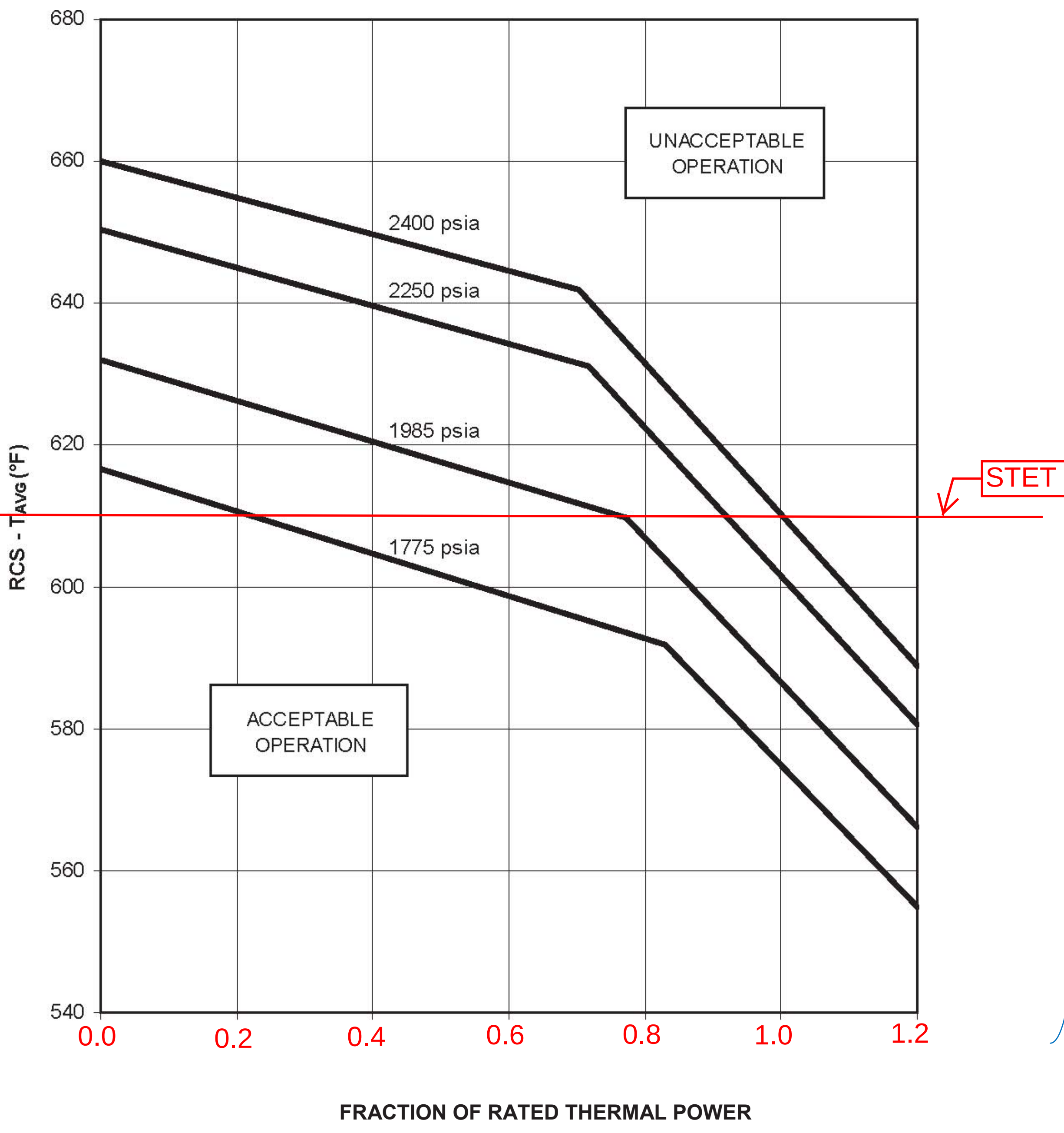
ITS A01 ITS Chapter 2.0

STET

Figure 2.1.1-1 ~~Figure 2.1-1 Reactor Core Safety Limit-Four Loops in Operation~~



SII ~~LA01~~ SII

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DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

Not Used

SII

LA01 ~~*(Type 6 – Removal of Cycle – Specific Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 2.1.1 requires the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature~~ ~~(T_{avg})~~ ~~not to exceed the limits shown in Figure 2.1-1. ITS 2.1.1 states the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR. This changes the CTS by moving limits that must be confirmed on a cycle specific bases to the COLR. The Reactor Core safety limits are retained in Technical Specification Chapter 2.0.~~

~~The removal of these cycle specific parameter limits from the Technical Specifications to the COLR and the retention of the limiting Safety Limits in the Technical Specifications is acceptable because the cycle specific limits are developed or utilized under NRC-approved methodologies that ensure the Safety Limits are met. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Safety Limits. NRC-approved Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for these parameters may be relocated to the COLR provided the SLs continue to appear in the Technical Specifications. The methodologies used to develop the parameters in the COLR were approved by the NRC in accordance with Generic Letter 88-16. Additionally, this change is acceptable because the removed information will be adequately controlled in the COLR~~

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

Not Used

~~under the requirements provided in ITS 5.6.3, "Core Operating Limits Report." ITS 5.6.3 ensures that the applicable limits of the safety analysis are met (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits). This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.~~

SII

LESS RESTRICTIVE CHANGES

None

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

2.0 SAFETY LIMITS (SLs)

2.1 **2.1 SLs**

2.1.1, 2.1.1 Applicability

2.1.1 Reactor Core SLs

SII

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits ~~specified in the COLR~~ shown in Figure 2.1.1-1; and the following SLs shall not be exceeded: 2

2.1.1.1 — 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ ~~[1.17 for the WRB-1/WRB-2 DNB correlations]~~ INSERT 1. 1

2.1.1.2 — 2.1.1.2 The ~~peak fuel~~ maximum local fuel pin centerline temperature shall be maintained ~~≤ [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup]~~ INSERT 2. 2 1

2.1.2, 2.1.2 Applicability

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2735~~]~~ psig. 1

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 ACTION — 2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION — 2.2.2 If SL 2.1.2 is violated:

2.1.2 ACTION — 2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION — 2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Insert 3

SII 2

1 **INSERT 1**

1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

1 **INSERT 2**

≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications

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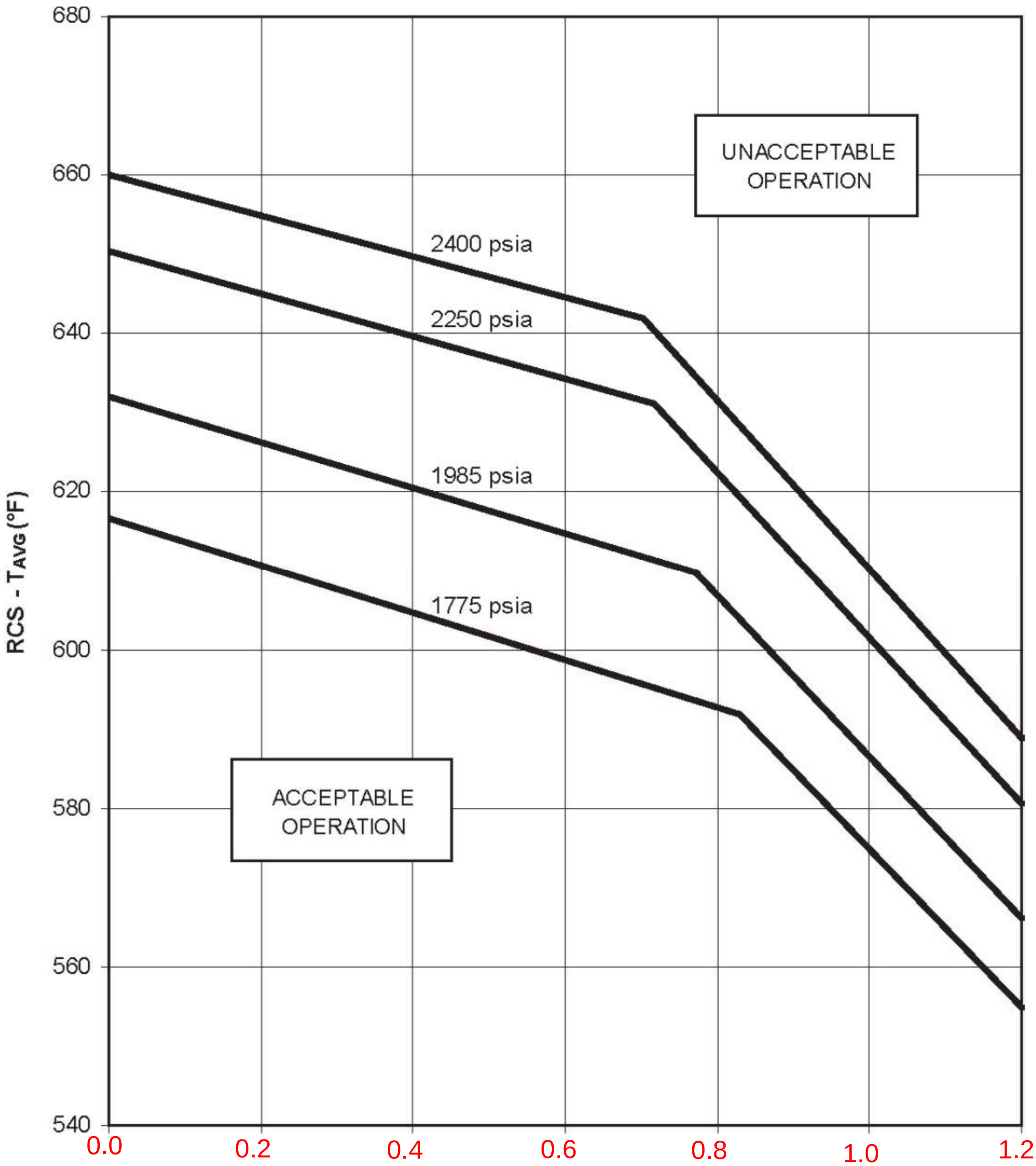
INSERT 3

Figure 2.1-1

Figure 2.1.1-1 Reactor Core Safety Limit Four Loops in Operation



FRACTION OF RATED THERMAL POWER

2.0 SAFETY LIMITS (SLs)

2.1 2.1 SLs

2.1.1, 2.1.1 Applicability 2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits ~~specified in the COLR~~ shown in Figure 2.1.1-1; and the following SLs shall not be exceeded: SII 2

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2.1.2, 2.1.2 Applicability 2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2735~~]~~ psig. 1

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 ACTION 2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION 2.2.2 If SL 2.1.2 is violated:

2.1.2 ACTION 2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION 2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

SII

Insert 3 2

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CTS 2.1.1

1 **INSERT 1**

1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

1 **INSERT 2**

≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications

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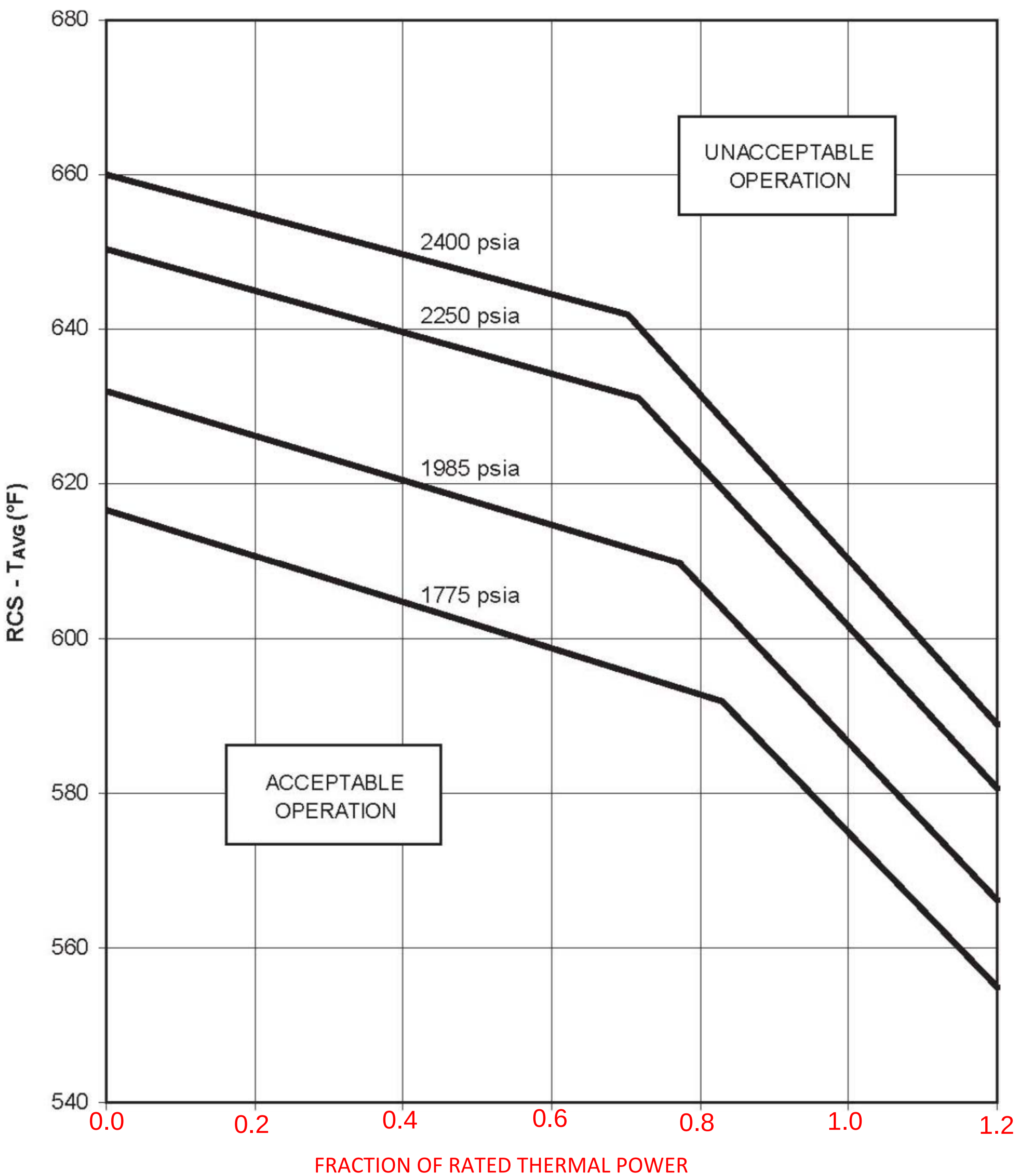
2.1.1

INSERT 3

Figure 2.1-1

Figure 2.1.1-1 Reactor Core Safety Limit Four Loops in Operation



JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

INSERT 1

The restrictions of this SL prevent overheating of the fuel ~~and~~ cladding~~, as well as possible~~ cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. 1

~~Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.~~

corresponding significant

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the ~~resultant sharp~~ reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

INSERT 2 1

INSERT 3 1

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

1 **INSERT 1**

(due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt(CFM)), either of which could result in

1 **INSERT 2**

from the outer surface of the cladding to the reactor coolant water

1 **INSERT 3**

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the value acceptance criteria in the safety analysis.

1 **INSERT 3 (cont)**

Figure 2.1.1-1

SII

The curves provided in ~~the COLR~~ show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

Figure 2.1.1-1

These lines are bounding for all fuel types. The curves provided in ~~the COLR~~ are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip set points are verified to be less than the limits defined by the safety limit lines provided in ~~the COLR~~ converted from power to delta-temperature and adjusted for uncertainty.

Figure 2.1.1-1

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (Delta I) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip set point is reduced by the values in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_2(\Delta I)$ function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_2(\Delta I)$ trip reset function, the Overpower-Delta Temperature trip set point is reduced by the values specified in the COLR to provide protection required by the core safety limits.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with ~~all~~ the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. 1

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the U FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

SII

SAFETY LIMITS

Figure 2.1.1-1 ~~The figure provided in the COLR~~ shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation. 1

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. U FSAR, Section ~~[~~7.2~~]~~. 1 2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure ~~coolant~~ boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. 3

stet CSS-005

The design pressure of the RCS is ~~2485 psig~~ 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). ~~1~~

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

BASES

APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs)~~,~~ ; 4

~~b. Steam line relief valve,~~ 4

~~c~~ b. Steam Dump System~~,~~ ; 5

~~d~~ c. Reactor Control System~~,~~ ; 4

~~e~~ d. Pressurizer Level Control System~~,~~ ; or 4

~~f~~ e. Pressurizer spray valve. 4

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~ is 120% of design pressure. 2
The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

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BASES

APPLICABILITY	SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.
SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28. 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971. 1 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000. 4. 10 CFR 100. 5. UFSAR, Section ~~[~~7.2~~]~~. 1 2 6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

INSERT 1

The restrictions of this SL prevent overheating of the fuel ~~and~~ cladding~~, as well as possible~~ cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. 1

~~Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.~~

corresponding significant

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the ~~resultant sharp~~ reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. INSERT 2 1

INSERT 3 1

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

1 **INSERT 1**

(due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt(CFM)), either of which could result in

1 **INSERT 2**

from the outer surface of the cladding to the reactor coolant water

1 **INSERT 3**

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the value acceptance criteria in the safety analysis.

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1

INSERT 3 (cont)

SII

Figure 2.1.1-1

The curves provided in ~~the COLR~~ show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

Figure 2.1.1-1

These lines are bounding for all fuel types. The curves provided in ~~the COLR~~ are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip set points are verified to be less than the limits defined by the safety limit lines provided in ~~the COLR~~ converted from power to delta-temperature and adjusted for uncertainty.

Figure 2.1.1-1

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (Delta I) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip set point is reduced by the values in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_2(\Delta I)$ function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_2(\Delta I)$ trip reset function, the Overpower-Delta Temperature trip set point is reduced by the values specified in the COLR to provide protection required by the core safety limits.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with ~~all~~ the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. 1

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the U FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

SII

SAFETY LIMITS

Figure 2.1.1-1 ~~The figure provided in the COLR~~ shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation. 1

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. U FSAR, Section ~~[~~7.2~~]~~. 1 2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure ~~coolant~~ boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. 3

stet CSS-005

The design pressure of the RCS is ~~2485 psig~~ ~~2500 psia~~. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). ~~1~~

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

BASES

APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs)~~,~~ ; 4

~~b. Steam line relief valve,~~ 4

b ~~c~~. Steam Dump System~~,~~ ; 5

c ~~d~~. Reactor Control System~~,~~ ; 4

d ~~e~~. Pressurizer Level Control System~~,~~ ; or 4

e ~~f~~. Pressurizer spray valve. 4

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~ is 120% of design pressure. 2
The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971. 1

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. UFSAR, Section ~~[~~7.2~~]~~. 1 2

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0 BASES, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Typographical/grammatical error corrected.

4. The steam line relief valves are removed from the list of items that have no credit taken for operation. The steam line safety valves are credited with protecting the Reactor Coolant System and the steam generators against overpressure for all load losses. Additionally, the subsequent items have been renumbered.

5. The punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

There are no specific No Significant Hazards Considerations for this Specification.

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ENCLOSURE 2

VOLUME 5

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.0
LCO AND SR APPLICABILITY

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Section 3.0, LCO AND SR APPLICABILITY

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

ITS A01 ITS Section 3.0

.0 (LCO)

3/~~4~~ LIMITING CONDITION~~S~~ FOR OPEATION ~~AND SURVEILLANCE REQUIREMENTS~~

~~3/4.0~~ APPLICABILITY

~~LIMITING CONDITION FOR OPERATION~~

A01

LCOs shall be met

LCO 3.0.1 3.0.1 ~~Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required~~ during the ~~OPERATIONAL~~ MODES or other conditions specified ~~therein;~~ except ~~that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met and~~ as provided in LCO ~~3.0.7~~.

LCO 3.0.2, LCO 3.0.7, 3.0.8, and LCO 3.0.9 in the Applicability, A02

LCO 3.0.2 3.0.2 ~~Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Conditions for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.~~

INSERT 1 A02 A03

an LCO and S

LCO 3.0.3 3.0.3 When ~~a Limiting Condition for Operation~~ is not met~~, except as provided in~~ the associated ACTION ~~requirements,~~ within one hour action shall be initiated to place the unit ~~in a MODE in which the Specification does not apply by placing it~~, as applicable, in:

INSERT 2 A04 A05

MODE 3 7

1. ~~At least~~ ~~HOT STANDBY~~ within ~~the next 6~~ hours,
2. ~~At least~~ ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours, and
3. ~~At least~~ ~~COLD SHUTDOWN~~ within ~~the subsequent 24~~ hours.

MODE 4 13 MODE 5 37

INSERT 3 A06

Where corrective measures are completed that permit operation ~~under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation~~. Exceptions to ~~these requirements~~ are stated in the individual Specifications.

this specification A01

LCO 3.0.4 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications~~,~~ ; or

A01

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this Specification. Unless both

See ITS 3.8.1

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A02 A03 **INSERT 1**

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

INSERT 2

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. A04 A01

A06 **INSERT 3**

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

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A01

ITS ITS Section 3.0

APPLICABILITY

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 (Continued)

conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in MODES 5 or 6.

See ITS 3.8.1

LCO 3.0.5 3.0.6 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO ~~3.0.1 and~~ 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. INSERT 4 INSERT 5

A07 A08 A09

LCO 3.0.8 3.0.7 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met. INSERT 6

L01

3.0 SURVEILLANCE REQUIREMENT~~S~~ (SR) APPLICABILITY

SR 3.0.1 4.0.1 ~~Surveillance Requirements~~ SRs shall be met during the MODES or other specified conditions in the Applicability for individual ~~Limiting Condition for Operation~~ LCOs, unless otherwise stated in the ~~individual Surveillance Requirement~~ SR. Failure to meet a Surveillance ~~Requirement~~, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the ~~Limiting Condition for Operation~~ LCO. Failure to perform a Surveillance within the specified ~~surveillance interval~~ Frequency shall be failure to meet the ~~Limiting Conditions for Operation~~ LCO except as provided in ~~Specification 4.0.3~~ SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

A01

SR 3.0.2 4.0.2 ~~Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.~~ INSERT 7

A10 L02 M01 A11

SR 3.0.3 4.0.3 If it is discovered that a Surveillance was not performed within its specified ~~surveillance interval (including the allowed extension per Specification 4.0.2)~~ Frequency, then compliance with the requirement to declare the ~~Limiting Condition for Operation~~ LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified ~~surveillance interval~~ Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

A01 A01

A08 **INSERT 4**

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

A09 **INSERT 5**

LCO 3.0.7 Test Exception LCO 3.1.8, PHYSICS TEST Exceptions – MODE 2" allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

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L01 **INSERT 6**

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

INSERT 7

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. A10

For Frequencies specified as "once," the above interval extension does not apply. M01

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L02

Exceptions to this Specification are stated in the individual Specifications. A10

ITS A01 ITS Section 3.0

SURVEILLANCE REQUIREMENTS (Continued)

4.0.3 (Continued)

SR 3.0.3 If the Surveillance is not performed within the delay period, the ~~Limiting Condition for Operation~~ LCO must immediately be declared not met, and the applicable ~~ACTION~~ Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the ~~Limiting Condition for Operation~~ LCO must immediately be declared not met, and the applicable ~~ACTION~~ Condition(s) must be entered. A01

SR 3.0.4 4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be as follows:

See ITS 5.5

Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;

b. The provisions of SR 4.0.2 are applicable to the frequencies for performing inservice inspection activities;

c. Inspection of each reactor coolant pump flywheel per the recommendation of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 or in lieu of Position c.4.b(1) and c.4.b(2), a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals (the provisions of SR 4.0.2 are not applicable); and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirement of any TS.

Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Testing Frequencies applicable to the ASME OM Code and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

See ITS 5.5

c. The provisions of SR 4.0.2 are applicable to the above required Frequencies and other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

d. The provisions of SR 4.0.3 are applicable to inservice testing and activities; and

e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

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ITS A01 ITS Section 3.0

.0 (LCO)

3/~~4~~ LIMITING CONDITION~~S~~ FOR OPERATION ~~AND SURVEILLANCE REQUIREMENTS~~ A01

~~3/4.0~~ APPLICABILITY

~~LIMITING CONDITION FOR OPERATION~~

LCOs shall be met

LCO 3.0.1 3.0.1 ~~Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required~~ during the ~~OPERATIONAL~~ MODES or other conditions specified ~~therein;~~ except ~~that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met and~~ as provided in LCO ~~3.0.7~~. A02

in the Applicability, LCO 3.0.2, LCO 3.0.7, 3.0.8, and LCO 3.0.9

LCO 3.0.2 3.0.2 ~~Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Conditions for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.~~ INSERT 1 A02 A03

an LCO and S

LCO 3.0.3 3.0.3 When ~~a Limiting Condition for Operation~~ is not met, ~~except as provided in~~ the associated ACTION ~~requirements,~~ within one hour action shall be initiated to place the unit ~~in a MODE in which the Specification does not apply by placing it~~, as applicable, in: INSERT 2 A04

7, 13, 37

1. ~~At least HOT STANDBY~~ within ~~the next 6~~ hours, MODE 3
2. ~~At least HOT SHUTDOWN~~ within ~~the following 6~~ hours, and MODE 4
3. ~~At least COLD SHUTDOWN~~ within ~~the subsequent 24~~ hours. MODE 5

A05

INSERT 3

Where corrective measures are completed that permit operation ~~under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation.~~ Exceptions to ~~these requirements~~ are stated in the individual Specifications. A06 A01

this specification

LCO 3.0.4 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications~~,~~ ; or A01

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this Specification. Unless both See ITS 3.8.1

A02 A03 **INSERT 1**

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

INSERT 2

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, A04
the unit shall be placed in a MODE or other specified condition in which the LCO is not A01
applicable.

A06 **INSERT 3**

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

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ITS A01 ITS Section 3.0

APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.5 (Continued)

conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in:

1. At least HOT STANDBY within the next 6 hours
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in MODES 5 or 6.

See ITS 3.8.1

LCO 3.0.5 3.0.6 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO ~~3.0.1 and~~ 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. INSERT 4 INSERT 5

A07 A08 A09

LCO 3.0.8 3.0.7 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met. INSERT 6

L01

3.0 SURVEILLANCE REQUIREMENT~~S~~ (SR) APPLICABILITY

SR 3.0.1 4.0.1 SRs ~~Surveillance Requirements~~ shall be met during the MODES or other specified conditions in the Applicability for individual LCOs ~~Limiting Condition for Operation~~, unless otherwise stated in the SR ~~individual Surveillance Requirement~~. Failure to meet a Surveillance ~~Requirement~~, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO ~~Limiting Condition for Operation~~. Failure to perform a Surveillance within the specified Frequency ~~surveillance interval~~ shall be failure to meet the LCO ~~Limiting Conditions for Operation~~ except as provided in SR 3.0.3 ~~Specification 4.0.3~~. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

A01

SR 3.0.2 4.0.2 ~~Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.~~ INSERT 7

A10 L02 M01 A01 A11 A01

SR 3.0.3 4.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency ~~surveillance interval~~ ~~(including the allowed extension per Specification 4.0.2)~~, then compliance with the requirement to declare the LCO ~~Limiting Condition for Operation~~ not met may be delayed, from the time of discovery,

A08 **INSERT 4**

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

A09 **INSERT 5**

LCO 3.0.7 Test Exception LCO 3.1.8, PHYSICS TEST Exceptions – MODE 2" allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

L01

INSERT 6

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

INSERT 7

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. A10

For Frequencies specified as "once," the above interval extension does not apply. M01

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L02

Exceptions to this Specification are stated in the individual Specifications. A10

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ITS A01 ITS Section 3.0

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.3 (Continued)

Frequency

up to 24 hours or up to the limit of the specified ~~surveillance interval~~, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. A01

LCO

SR 3.0.3 If the Surveillance is not performed within the delay period, the ~~Limiting Condition for Operation~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the ~~Limiting Condition for Operation~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be entered. A01

Condition LCO Condition

SR 3.0.4 4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be as follows:

Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;

b. The provisions of SR 4.0.2 are applicable to the frequencies for performing inservice inspection activities;

c. Inspection of each reactor coolant pump flywheel per the recommendation of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 or in lieu of Position c.4.b(1) and c.4.b(2), a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals (the provisions of SR 4.0.2 are not applicable); and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirement of any TS.

Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

See ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies applicable to the ASME OM Code and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

See ITS 5.5

c. The provisions of SR 4.0.2 are applicable to the above required Frequencies and other normal and accelerated frequencies specified as 2 years or less in the Inservice Test Program for performing inservice testing activities;

d. The provisions of SR 4.0.3 are applicable to inservice testing and activities; and

e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 Compliance with the Limiting Conditions for Operation (LCO) contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met and as provided in LCO 3.0.7." ITS LCO 3.0.1 states, "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9." This results in several changes to the CTS.

- Certain phrases are revised to be consistent with the equivalent phrase used in the ITS. Specifically, "OPERATIONAL MODES or other conditions specified therein" is changed to "MODES or other specified conditions in the Applicability" to be consistent with the ITS definition of MODE and the terminology used in the ITS.

  These changes are acceptable because they result in no change in the intent or application of the Technical Specifications, but merely reflect editorial preferences used in the ITS.

- The phrase "Compliance with the LIMITING CONDITIONS FOR OPERATION contained in the succeeding Specifications is required" is replaced with "LCOs shall be met." This change is made consistent with the ITS.

  This change is acceptable because it is an editorial change that does not change the intent of the requirements.

- The phrase "except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met" is moved from CTS 3.0.1 to ITS LCO 3.0.2, which states in part, "Upon discovery or failure to meet an LCO, the Required Actions of the associated Conditions shall be met."

  The change is acceptable because moving this information within the Technical Specifications results in no change in the intent or application of ACTIONS.

DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

- The phrase "except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9" is added in ITS LCO 3.0.1. ITS LCO 3.0.2 describes the appropriate actions to be taken when ITS LCO 3.0.1 is not met. ITS LCO 3.0.7 describes Test Exceptions LCOs, which are exceptions to other LCOs. ITS LCO 3.0.8 addresses snubber inoperabilities, which is also an exception to other LCOs. ITS LCO 3.0.9 addresses barrier inoperabilities which is also an exception to other LCOs. Changes resulting from the incorporation of ITS LCO 3.0.9 are discussed in Discussion of Change (DOC) L01.

This change is acceptable because adding the exceptions for ITS LCO 3.0.2, LCO 3.0.6, LCO 3.0.7, and LCO 3.0.9 prevent a conflict within the Applicability section. This addition is needed for consistency in the ITS requirements and does not change the intent or application of the Technical Specifications. Furthermore, changing the CTS LCO 3.0.7 to ITS LCO 3.0.8 does not change the intent of the snubber exception.

A03 CTS 3.0.2 states, "Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Conditions for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required." ITS LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated." This results in several changes to the CTS.

- CTS 3.0.2 is revised to include an exception for ITS LCO 3.0.6. ITS LCO 3.0.6 is a new allowance that takes exception to the ITS LCO 3.0.2 requirement to take the associated ACTION requirements when a LIMITING CONDITION FOR OPERATION is not met. This exception is included in ITS LCO 3.0.2 to avoid conflict between the applicability requirements.

  This change is acceptable because it includes a reference to a new item in the ITS and results in no change to the CTS. Changes resulting from the incorporation of ITS LCO 3.0.6 are discussed in DOC A07.

- The second sentence of CTS 3.0.2 states, "If the LIMITING CONDITIONS FOR OPERATION is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required." The sentence is changed, in ITS LCO 3.0.2, to state "If the LCO is not met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated."

  This change is acceptable because, while worded differently, both the CTS and ITS state that ACTIONS do not have to be completed once the LCO is met or is no longer applicable. ITS LCO 3.0.2 also adds the phrase "unless otherwise stated." There are some ITS ACTIONS that must be completed,

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DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

even if the LCO is met or is no longer applicable. This change is acceptable because it reflects a new feature in the ITS which does not exist in the CTS. The technical aspects of these changes are discussed in the appropriate ITS sections.

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

A04 CTS 3.0.3, in part, is applicable "When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements." ITS LCO 3.0.3 expands those applicability requirements so that the requirement is applicable "When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS." This changes the CTS to add two new applicability conditions.

- ITS LCO 3.0.3 is applicable when the LCO is not met and there is no applicable ACTION to be taken.

  This change is acceptable because it is consistent with the current understanding and application of CTS 3.0.3.

- ITS LCO 3.0.3 is applicable when directed by the associated ACTIONS. The CTS and the ITS contain such requirements. Any technical changes related to directing LCO 3.0.3 entry in an ACTION will be discussed in the affected Technical Specifications.

  This change is acceptable because it is consistent with the current understanding and application of CTS 3.0.3.

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

A05 CTS 3.0.3, in part, states that within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in: Hot Standby within the next 6 hours, Hot Shutdown within the following 6 hours, and at least Cold Shutdown within the subsequent 24 hours. ITS LCO 3.0.3 states that action shall be initiated within 1 hour to place the unit, as applicable, in MODE 3 within 7 hours, MODE 4 within 13 hours, and MODE 5 within 37 hours. This changes the CTS by using the sum of the times (i.e., the ITS Completion Time of 37 hours to enter MODE 5 is the same as the sum of the CTS allowance of 1 hour, 6 hours, 6 hours, and 24 hours) instead of sequential times (i.e., each time is measured from the completion of the previous step). The stated times in CTS 3.0.3 and ITS 3.0.3 are listed below:

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DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

Mode	Title	CTS Time to Enter Mode	ITS Time to Enter Mode
--	(Current Mode)	1 hour to begin action	1 hour to begin action
3	Hot Standby	within the next 6 hours	7 hours
4	Hot Shutdown	within the following 6 hours	13 hours
5	Cold Shutdown	within the subsequent 24 hours	37 hours

The purpose of CTS 3.0.3 is to establish the shutdown requirements that must be implemented when an LCO is not met and the condition is not specifically addressed in the associated ACTION requirements. The delineated time limit allows the unit to be placed in a safe shutdown MODE when the plant cannot be maintained within the limits for safe operation. The time limit, specified in CTS 3.0.3 to reach the lower MODES of operation, permits the shutdown to proceed in a controlled manner that is well within the specified maximum cooldown rate. Furthermore, the time limit is within the cooldown capabilities of the plant assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies. In the CTS, this is accomplished by allowing a total of 37 hours for the plant to be in Cold Shutdown when a shutdown is required during the MODE of Operation. In the absence of specific guidance within the CTS, current SQN practice if the unit is in a lower MODE of Operation and a CTS 3.0.3 shutdown is required, is to apply the time limit for reaching the lower MODE of operation (i.e., each time limit is measured from the time the previous MODE is reached). In the ITS, the time limits for ITS LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower mode of operation when an ITS LCO 3.0.3 shutdown is required, the time limit for reaching the next lower MODE applies (i.e., if the plant is in MODE 3, 13 hours is allowed to reach MODE 4). ITS 3.0 Bases gives a detailed discussion on the use of applying the allowed outage times when the unit is in a lower MODE when ITS 3.0.3 is entered. This is further explained, with examples, in the discussion of Section 1.3, "Completion Times." This change is acceptable because ITS and CTS both allow 37 hours to reach MODE 5 from power operation. In addition, the CTS 3.0.3 statement " within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply" has been editorially reworded in ITS LCO 3.0.3 to "the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. ACTION shall be initiated within 1 hour to place the unit..." These changes are considered changes to the CTS presentation. These changes are designated as administrative as they apply rules of usage established by ITS without resulting in technical changes to the Technical Specifications.

A06 CTS 3.0.3 states "Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in

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DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation." ITS LCO 3.0.3 states "Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is applicable in MODES 1, 2, 3, and 4."

This change is acceptable because the changes to CTS 3.0.3 are editorial. Both the CTS and ITS state that LCO 3.0.3 can be exited if the LCO which led to the entry into LCO 3.0.3 is met, or if one of the ACTIONS of that LCO is applicable. The CTS requirement also specifies that the time to complete the ACTIONS in the LCO is based on the initial failure to meet the LCO. Reentering the LCO after exiting LCO 3.0.3 does not reset the ACTION statement time requirements. This information is not explicitly stated in ITS LCO 3.0.3 but is true under the multiple condition entry concept of the ITS. In addition, the sentence "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4" is added to ITS LCO 3.0.3. CTS 3.0.3 and ITS LCO 3.0.3 require the unit to be placed only as low as COLD SHUTDOWN (MODE 5). Once the unit is in MODE 5, there are no further requirements. Thus, CTS 3.0.3 and ITS LCO 3.0.3 are effectively only applicable in MODES 1, 2, 3, and 4, and the addition of the sentence merely reflects editorial preferences used in the ITS.

These changes are designated as administrative because there is no change in the intent or application of the CTS 3.0.3 requirements.

A07 CTS 3.0.6 has a statement that CTS 3.0.6 is an exception to both CTS 3.0.1 and CTS 3.0.2. ITS LCO 3.0.5 includes only a statement that ITS LCO 3.0.5 is an exception to LCO 3.0.2. The statement that ITS LCO 3.0.5 is an exception to LCO 3.0.1 is not included.

This change is acceptable since ITS LCO 3.0.5 does not modify ITS LCO 3.0.1. The ACTION requirements discussion that is in CTS 3.0.1 has been moved to ITS LCO 3.0.2 (i.e., it is not included in ITS LCO 3.0.1). This change is designated as administrative since it does not result in any technical change to the Technical Specifications.

A08 ITS LCO 3.0.6 is added to the CTS to provide guidance regarding the appropriate ACTIONS to be taken when a single inoperability (a support system) also results in the inoperability of one or more related systems (supported system(s)). ITS LCO 3.0.6 states "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in

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accordance with LCO 3.0.2." In the CTS, based on the intent and interpretation provided by the NRC over the years, there has been an ambiguous approach to the combined support/supported inoperability. Some of this history is summarized below:

- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Engineering and Projects) to Samuel E. Bryan (Assistant Director for Field Coordination) would indicate an intent/interpretation consistent with the proposed LCO 3.0.6, without the necessity of also requiring additional ACTIONS. That is, only the inoperable support system ACTIONS need be taken.

- Guidance provided by the NRC in their April 10, 1980, letter to all Licensees, regarding the definition of OPERABILITY and its impact as a support system on the remainder of the CTS, would indicate a similar philosophy of not taking ACTIONS for the inoperable supported equipment. However, in this case, additional actions (similar to the proposed Safety Function Determination Program actions) were addressed and required.

- Regulatory Issue Summary (RIS) 2005-20 and a reading of the CTS provide an interpretation that inoperability, even as a result of a Technical Specification support system inoperability, requires all associated ACTIONS to be taken.

- Certain CTS contain ACTIONS such as "Declare the {supported system} inoperable and take the ACTIONS of {its Specification}." In many cases, the supported system would likely already be considered inoperable. The implication of this presentation is that the ACTIONS of the inoperable supported system would not have been taken without the specific direction to do so.

Considering the history of misunderstandings in this area, the WOG ISTS, NUREG-1431, Rev. 4, was developed with Industry input and approval of the NRC to include LCO 3.0.6 and a new program, Specification 5.5.13, "Safety Function Determination Program (SFDP)." This change is acceptable since its function is to clarify existing ambiguities and to maintain actions within the realm of previous interpretations. This change is designated as administrative because it does not technically change the Technical Specifications.

A09 ITS LCO 3.0.7 is added to the CTS. ITS LCO 3.0.7 states "Test Exception LCOs 3.1.8, "PHYSICS TEST Exceptions – MODE 2" allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications."

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This change is acceptable because the CTS contain test exception specifications that allow certain LCOs to not be met for the purpose of special tests and operations. However, the CTS does not contain the equivalent of ITS LCO 3.0.7. As a result, there could be confusion regarding which LCOs are applicable during special tests. LCO 3.0.7 was crafted to avoid that possible confusion. LCO 3.0.7 is consistent with the use and application of CTS test exception Specifications and does not provide any new restriction or allowance. This change is designated as administrative because it does not technically change the Technical Specifications.

A10 CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." ITS SR 3.0.2 states, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This results in several changes to the CTS.

- ITS SR 3.0.2 adds to the CTS "For Frequencies specified as "once," the above interval extension does not apply." This is described in DOC M01.

- ITS SR 3.0.2 adds to the CTS "If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." This is covered by DOC L02.

- CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." ITS SR 3.0.2 states, in part, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency." This change is being made to be consistent with the ITS terminology and to clarify the concept of the specified SR Frequency being met.

  This change is acceptable since it does not change the intent of the requirements.

- ITS SR 3.0.2 is more specific regarding the state of the Frequency by stating, "as measured from the previous performance or as measured from the time a specified condition of the Frequency is met." This direction is consistent with the current use and application of the Technical Specifications.

  This change is acceptable because the ITS intent is the same as the CTS requirement.

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- ITS SR 3.0.2 adds to the CTS "Exceptions to this Specification are stated in the individual Specifications."

  This change is acceptable because it reflects practices used in the ITS that are not used in the CTS. Any changes to a Technical Specification, by inclusion of such an exception, will be addressed in the affected Technical Specification.

The changes, except as discussed in DOC M01 and DOC L02, are designated as administrative because they reflect presentation and usage rules of the ITS without making technical changes to the Technical Specifications.

A11 CTS 4.0.3 states, in part, "If it is discovered that a Surveillance was not performed within its specified surveillance interval (including the allowed extension per Specification 4.0.2), then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater." ITS SR 3.0.3 contains a similar requirement, but excludes the statement "(including the allowed extension per Specification 4.0.2)." This changes the CTS by not including the statement "(including the allowed extension per Specification 4.0.2)."

This change is acceptable because the statement in CTS 4.0.2 "(including the allowed extension per Specification 4.0.2)" is not needed to be repeated in ITS SR 3.0.3. ITS SR 3.0.2 allows a Surveillance to be performed within 1.25 times the interval specified in the Frequency. Therefore, there is no need to repeat in ITS SR 3.0.3 the allowance that is granted in ITS SR 3.0.2. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

MORE RESTRICTIVE CHANGES

M01 CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." ITS SR 3.0.2 states, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding "For Frequencies specified as "once," the above interval extension does not apply." The remaining changes to CTS 4.0.2 are discussed in DOC A10 and DOC L02.

The purpose of the 1.25 extension allowance to Surveillance Frequencies is to allow for flexibility in scheduling tests. This change is acceptable because Frequencies specified as "once" are typically condition-based Surveillances in

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which the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Frequency without extension in order to avoid operation in unacceptable conditions. This change is designated as more restrictive because an allowance to extend Frequencies by 25 percent is eliminated for some Surveillances.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 CTS Section 3.0 does not contain an allowance when barriers cannot support their support function. The proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.9. The addition of LCO 3.0.9 to the CTS is to address barriers which cannot perform their related support function for Technical Specification systems. ITS LCO 3.0.9 allows barriers to be able to not perform their safety function for up to 30 days before declaring the supported system inoperable. Furthermore, due to this addition, an allowance is also needed in LCO 3.0.1. This allowance has been added.

Barriers are defined as doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, which are designed to provide for the performance of the safety function for the Technical Specification system after the occurrence of one or more initiating events.

The barrier which cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and the associated industry guidance (NUMARC 93-01, Revision 3). This provision is applicable whether the barrier is affected due to planned maintenance or due to a discovered condition. Should the risk assessment and risk management actions for a specific plant configuration or emergent condition not support the 30 day allowed time, the Maintenance Rule risk management determined allowed time and actions must be implemented or the supported system’s LCO be considered not met.

Application of LCO 3.0.9 is dependent on the OPERABILITY of at least one train or subsystem of the supported Technical Specification system and the system's ability to mitigate the consequences of the specified initiating events. However, during the 30 day period allowed by LCO 3.0.9, there exists the possibility that the train or subsystem required to be OPERABLE will unexpectedly become inoperable. Absent any further consideration, this would likely result in both trains of a Technical Specification required system being declared inoperable

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(i.e., the train supported by the barriers to which LCO 3.0.9 was being applied and the emergent condition of the inoperable train). This would likely result in entering LCO 3.0.3 and a rapid plant shutdown. While this scenario is of low likelihood, it is of very high consequence to the licensee and, therefore, should be avoided unless necessary to avoid an actual plant risk. As a result, LCO 3.0.9 contains a provision which addresses the emergent condition of the required OPERABLE train or subsystem becoming inoperable while LCO 3.0.9 is being used. LCO 3.0.9 provides 24 hours to either restore the inoperable train or subsystem or to cease relying on the provisions of LCO 3.0.9 to consider the train or subsystem supported by the affected barrier(s) OPERABLE. This 24 hour period is not based on a generic risk evaluation, as it would be difficult to perform such an analysis in a generic fashion. Rather, plant risk during this 24 hour allowance is managed using the contemporaneous risk assessment and management required by 10 CFR 50.65(a)(4) and recognizes the unquantified advantage to plant safety of avoiding a plant shutdown with the associated transition risk.

A risk impact of the 30 day allowance for barriers was performed. All Sequoyah initiating events are located on the table depicted in TSTF-427 OR Sequoyah has evaluated the use of LCO 3.0.9 for a barrier protecting against an initiating event not on the table located in TSTF-427 and calculated the frequency ranges within the ranges in the table so the above analysis is applicable for those initiators. Therefore, LCO 3.0.9 can be utilized when inoperable barriers affect Systems, Structures, or Components (SSCs).

Insert 1 CSS-002

L02 CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." ITS SR 3.0.2 states, " The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, " If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." The remaining changes to CTS 4.0.2 are discussed in DOC A10 and DOC M01.

This change is acceptable because the 25 percent Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time with extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because addition time is provided to perform some periodic Required Actions.

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Insert 1

SQN is adopting TSTF-427, Revision 2, as incorporated in NUREG-1431, Revision 4, with no deviations from Specification LCO 3.0.9. TVA has reviewed the TSTF-427 documentation and the technical justifications presented in the model application safety evaluation prepared by the NRC staff and find that the technical justifications presented are applicable to SQN Units 1 and 2. In addition, TVA will be adopting the LCO 3.0.9 Bases, as indicated in the ITS conversion submittal. The only deviations to the LCO 3.0.9 Bases have been made for clarity and are justified in the Bases Justification for Deviations. CSS-002

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

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1 LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

3.0.2 LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

3.0.3 LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours~~,~~ ;

b. MODE 4 within 13 hours~~,~~ ; and 2

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0.4 LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications~~,~~ ; or 2

3.0 LCO Applicability

3.0.4 LCO 3.0.4 (continued)

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.6 LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

DOC A08 LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.~~15~~ 13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. 4

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

DOC A09 LCO 3.0.7 Test Exception LCO~~s~~ ~~[~~3.1.8 ~~and 3.4.19]~~ , "PHYSICS TEST Exceptions – MODE 2," allow s specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications. 3

3.0 LCO Applicability

3.0.7 LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

DOC L01 LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

4.0.3 SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

4.0.4 SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

3.0 SR Applicability

4.0.4 SR 3.0.4 (continued)

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1 **LCO 3.0.1** LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

3.0.2 **LCO 3.0.2** Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

3.0.3 **LCO 3.0.3** When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours~~,~~;

b. MODE 4 within 13 hours~~,~~; and 2

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0.4 **LCO 3.0.4** When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications~~,~~; or 2

3.0 LCO Applicability

3.0.4 LCO 3.0.4 (continued)

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.6 LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

DOC A08 LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.~~15~~ 13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. 4

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

DOC A09 LCO 3.0.7 Test Exception LCO~~s~~ ~~[~~3.1.8 , "PHYSICS TEST Exceptions – MODE 2," ~~and 3.4.19]~~ allow s specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications. 3

3.0 LCO Applicability

3.0.7 LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

DOC L01 LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1	SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
4.0.2	SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
4.0.3	SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
4.0.4	SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

3.0 SR Applicability

4.0.4 SR 3.0.4 (continued)

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.0, LCO AND SR APPLICABILITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes were made to use correct punctuation, typographical errors, or to make other corrections consistent with the Writers Guide for Improved Technical Specifications, TSTF-GG-05-01.

3. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

4. Changes were made to reflect changes made to the Specification.

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

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification ; ~~and~~ 2

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies ; or 2

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met~~,~~ ; 2

b. A Condition exists for which the Required Actions have now been performed~~,~~ ; or 2

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching

BASES

LCO 3.0.3 (continued)

MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Whenever

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.13 ~~15~~, "Spent Fuel ~~Storage~~ Pool Water Level." LCO 3.7.13 ~~15~~ has an Applicability of ~~"During movement of~~ irradiated fuel assemblies are in the spent fuel ~~storage~~ pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 ~~15~~ are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Actions of LCO 3.7.13 ~~15~~ of "Suspend all movement of ~~irradiated~~ fuel assemblies in the fuel ~~storage pool~~ spent fuel pool areas and crane operations with loads" and "Restore spent fuel pool water level to within limit" are ~~is~~ the appropriate Required Actions to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities ~~to~~ be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability. 3

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., ~~[~~Containment Air Temperature, Containment Pressure, ~~MCPR,~~ and Moderator Temperature Coefficient~~]~~), and may be applied to other Specifications based on NRC plant specific approval. 4

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

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BASES

LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service ; or 2

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

13

Specification 5.5.~~15~~, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. 5

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BASES

LCO 3.0.6 (continued)

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1)~~,~~ ; 2

b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2)~~,~~ ; or 2

c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

BASES

LCO 3.0.6 (continued)

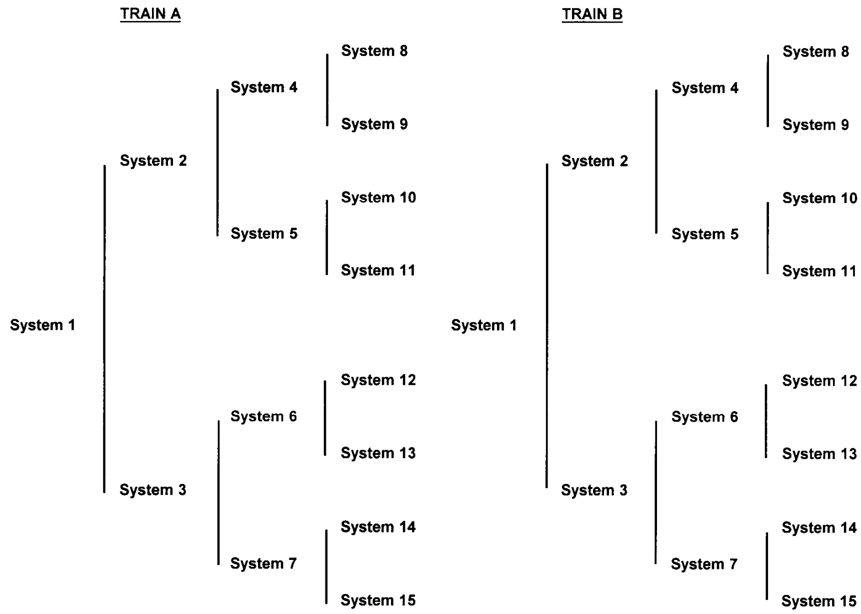

Figure B 3.0-1
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction

BASES

LCO 3.0.6 (continued)

source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO~~s~~ ~~[~~3.1.8 , "PHYSICS TEST Exceptions – MODE 2," ~~and 3.4.19]~~ allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect. 4

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

BASES

LCO 3.0.8 (continued)

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

BASES

~~-----REVIEWER'S NOTE-----~~
~~Adoption of LCO 3.0.9 requires the licensee to make the following commitments:~~

~~1. [LICENSEE] commits to the guidance of NUMARC 93-01, Revision 3, Section 11, which provides guidance and details on the assessment and management of risk during maintenance.~~ 6

~~2. [LICENSEE] commits to the guidance of NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006.~~
~~-----~~

LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely ~~due to~~ because required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that ~~as~~ required barriers are ~~found or are otherwise made unavailable, they are~~ ~~discovered~~ restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment. 3 CSS-004 3

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers ~~which~~ that are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001). 3

BASES

LCO 3.0.9 (continued)

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

~~-----REVIEWER'S NOTE-----~~
~~LCO 3.0.9 may be expanded to other initiating event categories provided plant-specific analysis demonstrates that the frequency of the additional initiating events is bounded by the generic analysis or if plant-specific approval is obtained from the NRC.~~ 6
~~-----~~

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train

BASES

LCO 3.0.9 (continued)

protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.

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SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures ~~>~~ greater than 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing. 3

b. High pressure safety injection (~~HPI~~ SI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with ~~HPI~~ SI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing. 1 1

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25~~%~~ percent extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). 3

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When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25 percent extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted. 7

The 25~~%~~ percent extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 3

BASES

SR 3.0.2 (continued)

percent

25~~%~~ extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations. As stated in SR 3.0.2, the 25 percent ~~%~~ extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25 percent ~~%~~ extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25 percent ~~%~~ extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner. 3

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

SII

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification). 7

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

BASES

SR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

BASES

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this

BASES

SR 3.0.4 (continued)

instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification ; and 2

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies ; or 2

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met~~,~~ ; 2

b. A Condition exists for which the Required Actions have now been performed~~,~~ ; or 2

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching

BASES

LCO 3.0.3 (continued)

MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Whenever

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.~~15~~ 13, "Spent Fuel ~~Storage~~ Pool Water Level." LCO 3.7.~~15~~ 13 has an Applicability of "~~During movement of~~ irradiated fuel assemblies in the spent fuel ~~storage~~ pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.~~15~~ 13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action s of LCO 3.7.~~15~~ 13 of "Suspend all movement of ~~irradiated~~ fuel assemblies in the ~~spent~~ stet fuel ~~storage~~ ~~pool~~ areas" and crane operations with loads and "Restore spent fuel pool water level to within limit" are ~~is~~ the appropriate Required Action s to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities ~~to~~ be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability. 3

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., ~~[~~Containment Air Temperature, Containment Pressure, ~~MCPR,~~ and Moderator Temperature Coefficient~~]~~), and may be applied to other Specifications based on NRC plant specific approval. 4

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

BASES

LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service ; or 2

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13 ~~15~~, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. 5

BASES

LCO 3.0.6 (continued)

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1)~~,~~ ; 2

b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2)~~,~~ ; or 2

c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

<u>EXAMPLE B 3.0.6-1</u>

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

<u>EXAMPLE B 3.0.6-2</u>

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

<u>EXAMPLE B 3.0.6-3</u>

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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BASES

LCO 3.0.6 (continued)

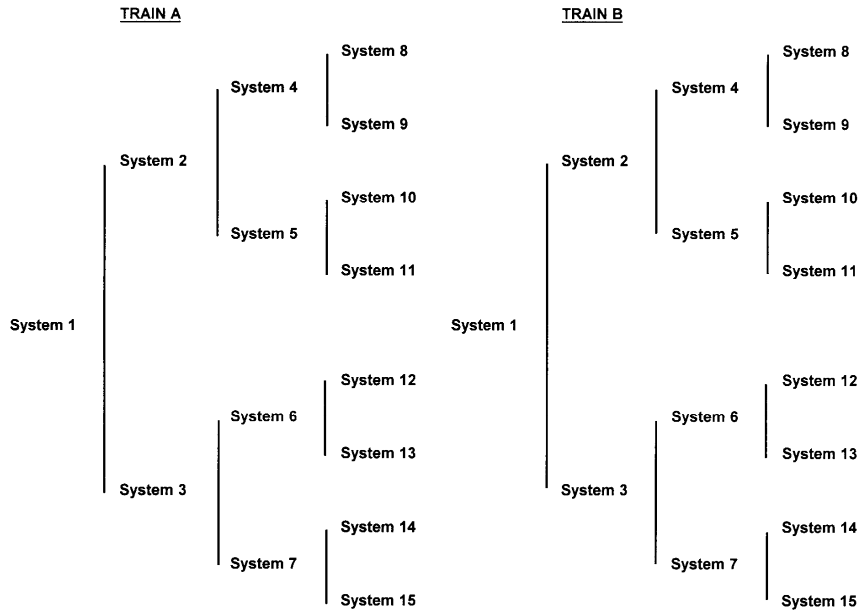

Figure B 3.0-1
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction

BASES

LCO 3.0.6 (continued)

source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO~~s~~ ~~[~~3.1.8 , "PHYSICS TEST Exceptions – MODE 2," ~~and 3.4.19]~~ allow s specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect. 4

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

BASES

LCO 3.0.8 (continued)

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

BASES

~~-----REVIEWER'S NOTE-----~~
~~Adoption of LCO 3.0.9 requires the licensee to make the following commitments:~~

~~1. [LICENSEE] commits to the guidance of NUMARC 93-01, Revision 3, Section 11, which provides guidance and details on the assessment and management of risk during maintenance.~~ 6

~~2. [LICENSEE] commits to the guidance of NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006.~~

LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely ~~due to~~ because required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that ~~as~~ required barriers are ~~found or are otherwise made unavailable, they are~~ ~~discovered~~ restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment. 3 CSS-004 3

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers ~~which~~ that are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001). 3

BASES

LCO 3.0.9 (continued)

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

~~-----REVIEWER'S NOTE-----~~
~~LCO 3.0.9 may be expanded to other initiating event categories provided plant-specific analysis demonstrates that the frequency of the additional initiating events is bounded by the generic analysis or if plant-specific approval is obtained from the NRC.~~ 6
~~-----~~

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train

BASES

LCO 3.0.9 (continued)

protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.

SII

7

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures ~~>~~ greater than 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing. 3

b. High pressure safety injection (~~HPI~~ SI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with ~~HPI~~ SI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing. 1 1

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SII

When a Section 5.5, “Programs and Manuals,” specification states that the provisions of SR 3.0.2 are applicable, a 25 percent extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted. 7

SR 3.0.2 permits a 25~~%~~ percent extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). 3

The 25~~%~~ percent extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 3

BASES

SR 3.0.2 (continued)

25~~%~~ percent extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations. As stated in SR 3.0.2, the 25~~%~~ percent extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25~~%~~ percent extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25~~%~~ percent extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner. 3

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

SII

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification). 7

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

BASES

SR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

BASES

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this

BASES

SR 3.0.4 (continued)

instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

JUSTIFICATION FOR DEVIATIONS
ITS 3.0 BASES, LCO AND SR APPLICABILITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. Changes have been made for clarity.

4. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

5. Changes have been made to reflect changes made to the Specification.

6. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

7. Information has been added to the ISTS 3.0 Bases to clarify when SR 3.0.2 and SR 3.0.3 are applicable to Chapter 5 Specifications. SII

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L01

The Tennessee Valley Authority (TVA) is converting Sequoyah to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, Rev. 4, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) Less Restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

CTS Section 3.0 does not contain an allowance when barriers cannot support their support function. The proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.9. The addition of LCO 3.0.9 to the CTS is to address barriers which cannot perform their related support function for Technical Specification systems. ITS LCO 3.0.9 allows barriers to be able to not perform their safety function for up to 30 days before declaring the supported system inoperable. Furthermore, due to this addition, an allowance is also needed in LCO 3.0.1. This allowance has been added.

Barriers are defined as doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, which are designed to provide for the performance of the safety function for the Technical Specification system after the occurrence of one or more initiating events.

The barrier which cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and the associated industry guidance (NUMARC 93-01, Revision 3). This provision is applicable whether the barrier is affected due to planned maintenance or due to a discovered condition. Should the risk assessment and risk management actions for a specific plant configuration or emergent condition not support the 30 day allowed time, the Maintenance Rule risk management determined allowed time and actions must be implemented or the supported system's LCO be considered not met.

Application of LCO 3.0.9 is dependent on the OPERABILITY of at least one train or subsystem of the supported Technical Specification system and the system's ability to mitigate the consequences of the specified initiating events. However, during the 30 day period allowed by LCO 3.0.9, there exists the possibility that the train or subsystem required to be OPERABLE will unexpectedly become inoperable. Absent any further consideration, this would likely result in both trains of a Technical Specification required system being declared inoperable (i.e., the train supported by the barriers to which LCO 3.0.9 was being applied and the emergent condition of the inoperable train). This would likely result in entering LCO 3.0.3 and a plant shutdown. While this scenario is of low likelihood, it is of very high consequence to the licensee and, therefore, should be avoided unless necessary to avoid an actual plant risk. As a result, LCO 3.0.9 contains a provision which addresses the emergent condition of the required OPERABLE train or subsystem becoming inoperable while LCO 3.0.9 is being used. LCO 3.0.9 provides 24 hours to either restore the inoperable train or subsystem or to cease relying on the provisions of LCO 3.0.9 to consider the train or subsystem supported by the affected barrier(s) OPERABLE. This 24 hour period is not based on a generic risk evaluation, as it would be difficult to perform such an analysis in a generic fashion. Rather, plant risk

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

during this 24 hour allowance is managed using the contemporaneous risk assessment and management required by 10 CFR 50.65(a)(4) and recognizes the unquantified advantage to plant safety of avoiding a plant shutdown with the associated transition risk.

A risk impact of the 30 day allowance for barriers was performed. All Sequoyah initiating events are located on the table depicted in TSTF-427 OR Sequoyah has evaluated the use of LCO 3.0.9 for a barrier protecting against an initiating event not on the table located in TSTF-427 and calculated the frequency ranges within the ranges in the table so the above analysis is applicable for those initiators. Therefore, LCO 3.0.9 can be utilized when inoperable barriers affect Systems, Structures, or Components (SSCs).

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   Barriers are not an initiator to any accident previously evaluated. The probability of an accident previously evaluated is not significantly increased. Barriers support the operation of equipment assumed to mitigate the effects of accidents previously evaluated. The proposed relaxation may only be applied to a single train or subsystem of a multiple train or subsystem Technical Specification system at a given time for a given category of initiating event, or to multiple trains or subsystems of a multiple train or subsystem Technical Specification system provided the affected barriers protect against different categories of initiating events. Therefore, for any given category of initiating event, the ability to perform the assumed safety function is preserved. The consequences of an accident occurring during the time allowed when barriers are not capable of performing their related support function are no different from the consequences of the same accident while relying on the Actions of the supported Technical Specification systems.

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

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

   Response: No.

   No new or different accidents result from using the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

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

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

   Response: No.

   The proposed change allows for a limited period of time in which barriers may be unable to perform their related support function without declaring the supported systems inoperable. A risk analysis has shown that this provision will not have a significant effect on plant risk. In addition, regulatory requirements in 10 CFR 50.65(a)(4) require risk assessment and risk management, which will ensure that plant risk is not significantly increased.

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

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

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L02

The Tennessee Valley Authority (TVA) is converting Sequoyah to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, Rev. 4, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) Less Restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." ITS SR 3.0.2 states, " The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, " If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." The remaining changes to CTS 4.0.2 are discussed in DOC A10 and DOC M01.

This change is acceptable because the 25 percent Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time with extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because addition time is provided to perform some periodic Required Actions.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25 percent. This change does not significantly affect the probability of an accident. The length of time between performance of Required Actions is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated are the same during the Completion Time or during any extension of the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

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

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

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25 percent. This change will not involve physically altering the plant (i.e., no new or different type of equipment will be installed). In addition, the change does not involve any new or revised operator actions.

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

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

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25 percent. The 25 percent extension allowance is provided for scheduling convenience and is not expected to have significant effect on the average time between Required Actions. As a result, the Required Action will continue to provide appropriate compensatory measures for the subject Condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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ENCLOSURE 2

VOLUME 6

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.1
REACTIVITY CONTROL SYSTEMS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.1.1, SHUTDOWN MARGIN (SDM)

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

ITS A01 ITS 3.1.1

3/4.1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~ A01

SHUTDOWN MARGIN ~~- T_{avg} Greater Than 200°F~~ A02

LIMITING CONDITION FOR OPERATION

within the limits specified in the COLR LA01

LCO 3.1.1 3.1.1.1 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.6% delta k/k for 4 loop operation~~. A04

k_{eff} < 1.0 A03

Applicability APPLICABILITY: MODES 1, 2~~*~~, 3, and 4. See ITS 3.1.6

ACTION: not within limits LA01

within 15 minutes L01

ACTION A With the SHUTDOWN MARGIN ~~less than 1.6% delta k/k, immediately~~ initiate and ~~continue~~ boration ~~at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored. L02

SURVEILLANCE REQUIREMENTS

within the limits specified in the COLR LA01

SR 3.1.1.1 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.6% delta k/k~~:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS 3.1.4 See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.6

~~*See Special Test Exception 3.10.1~~ A04

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ITS A01 ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. ~~Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.~~ L03

MODE 2 with k_{eff} < 1.0 M01

SR 3.1.1.1 e. When in MODES 3 or 4, ~~at least once per 24 hours by consideration of the following factors:~~

In accordance with the Surveillance Frequency Control Program LA02

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration.~~

LA03

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

See ITS 3.1.2

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ITS A01 ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN ~~- T_{avg} Less Than or Equal to 200°F~~ A02

LIMITING CONDITION FOR OPERATION

within the limits specified in the COLR LA01

LCO 3.1.1 3.1.1.2 The SHUTDOWN MARGIN shall ~~be greater than or equal to 1.0% delta k/k~~.

Applicability APPLICABILITY: MODE 5.

ACTION: not within limits LA01

within 15 minutes L01

ACTION A With the SHUTDOWN MARGIN ~~less than 1.0% delta k/k, immediately~~ initiate and ~~continue~~ boration ~~at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored. L02

SURVEILLANCE REQUIREMENTS

within the limits specified in the COLR LA01

SR 3.1.1.1 4.1.1.2 The SHUTDOWN MARGIN shall be determined to ~~be greater than or equal to 1.0% delta k/k~~:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS 3.1.4 See ITS Chapter 1.0

b. ~~At least once per 24 hours by consideration of the following factors:~~ In accordance with the Surveillance Frequency Control Program LA02

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~ LA03

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration.~~

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ITS

A01

ITS 3.1.1

3/4.1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~ A01

SHUTDOWN MARGIN ~~- $T_{avg} \geq 200°F$~~ A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 3.1.1.1 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.6% delta k/k for 4 loop operation~~ within the limits specified in the COLR. LA01 A04

Applicability APPLICABILITY: MODES 1, 2~~*~~ $k_{eff} < 1.0$, 3, and 4. A03 See ITS 3.1.6

ACTION:

ACTION A With the SHUTDOWN MARGIN ~~less than 1.6% delta k/k,~~ not within limits ~~immediately~~ within 15 minutes initiate and ~~continue~~ boration ~~at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored~~. LA01 L01 L02

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.6% delta k/k~~ within the limits specified in the COLR: LA01

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS 3.1.4 See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2, with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.6

d. ~~Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.~~ L03

~~* See Special Test Exception 3.10.1~~ A04

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ITS

A01

ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

MODE 2 with k_{eff} < 1.0

M01

SR 3.1.1.1 e. When in MODES 3 or 4, ~~at least once per 24 hours by consideration of the following factors:~~

In accordance with the Surveillance Frequency Control Program

LA02

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration~~.

LA03

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

See ITS 3.1.2

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ITS

A01

ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN ~~- T_{avg} Less Than or Equal to 200°F~~

A02

LIMITING CONDITION FOR OPERATION

within the limits specified in the COLR

LA01

LCO 3.1.1 3.1.1.2 The SHUTDOWN MARGIN shall ~~be greater than or equal to 1.0% delta k/k~~.

Applicability APPLICABILITY: MODE 5.

ACTION:

not within limits

LA01

within 15 minutes

L01

L02

ACTION A With the SHUTDOWN MARGIN ~~less than 1.0% delta k/k, immediately~~ initiate and ~~continue~~ boration ~~at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

within the limits specified in the COLR

LA01

SR 3.1.1.1 4.1.1.2 The SHUTDOWN MARGIN shall be determined to ~~be greater than or equal to 1.0% delta k/k~~:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

See ITS Chapter 1.0

b. ~~At least once per 24 hours by consideration of the following factors:~~

In accordance with the Surveillance Frequency Control Program

LA02

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration~~.

LA03

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.1.1.1 provides the SHUTDOWN MARGIN (SDM) requirement in MODES 1, 2, 3, and 4 (i.e., T_{avg} greater than 200°F). CTS 3.1.1.2 provides the SDM requirement in MODE 5 (i.e., T_{avg} less than or equal to 200°F). ITS 3.1.1 provides the SDM requirement in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. This changes the CTS by combining the SDM requirements in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. The change in Applicability for MODE 2 with $k_{eff} < 1.0$ is described in DOC A03.

This change is acceptable because the requirements have not changed. Combining the Specifications is an editorial change. Any technical changes resulting from this combination are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.1.1.1 provides the SDM requirement in MODES 1, 2, 3, and 4 (i.e., T_{avg} greater than 200°F). CTS 4.1.1.1.1 states, when in MODES 1 and 2 with $k_{eff} \geq 1.0$, verify the control bank withdrawal is within the limits of Specification 3.1.3.6. ITS 3.1.1 is Applicable in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. This changes the CTS by combining the SDM requirement in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. The change in Applicability for MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ is described in ITS 3.1.6 (Control Bank Insertion Limits).

The purpose of CTS 3.1.1.1 is to ensure that the SDM assumed in the accident analysis is available. When the reactor is critical, SDM is verified by ensuring the control rods are within the control rod insertion limits. ITS 3.1.1 Applicability Bases state in MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." This change is acceptable because the SDM requirements have not changed. Even though CTS 3.1.1.1 is applicable in MODES 1 and 2, the CTS Surveillances only require the verification that control rod bank withdrawal is within the control rod insertion limits. The ITS verifies SDM in MODES 1 and 2 by the rod insertion limits. Any changes to the rod insertion limit requirements are discussed in DOCs for those Specifications. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.1.1.1 Applicability is MODES 1, 2, 3, and 4 with a footnote (footnote *) for MODE 2 stating "See Special Test Exception 3.10.1." ITS 3.1.1 does not contain

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DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

the footnote or a reference to the Special Test Exception. This changes the CTS by not including footnote * in the ITS.

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.1.1.1.1.e requires SDM to be determined to be within its limits every 24 hours when in MODES 3 and 4. ITS SR 3.1.1.1 requires SDM to be determined to be within its limits in MODE 2 with k_{eff} < 1.0 and MODES 3 and 4. This changes the CTS by expanding the applicability of the Surveillance to include MODE 2 with k_{eff} < 1.0.

The purpose of CTS 4.1.1.1.1.e is to verify that sufficient SDM is available. CTS 4.1.1.1.1.b states that when the reactor is in MODE 1 and MODE 2 with k_{eff} ≥ 1.0, SDM is verified by determining that the control rods are above the rod insertion limits. In MODE 2 with k_{eff} < 1.0, CTS 4.1.1.1.1.c verifies SDM by determining that the control rods are above the rod insertion limits. However, no CTS Surveillance requires a periodic verification of SDM when in MODE 2 with k_{eff} < 1.0. This change is acceptable because the ITS requires a specific verification that the SDM is within the limit when in MODE 2 with k_{eff} < 1.0 on a periodic basis. This change is designated as more restrictive because it expands the conditions under which a Surveillance must be performed.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.1.1.1, CTS 3.1.1.1 ACTION and CTS 4.1.1.1.1 require the SDM to be greater than or equal to 1.6% delta k/k when in MODES 1, 2, 3, and 4. CTS 3.1.1.2, CTS 3.1.1.2 ACTION and CTS 4.1.1.2.1 require the SDM to be greater than or equal to 1.0% delta k/k when in MODE 5. ITS LCO 3.1.1 requires the SDM to be within the limits specified in the COLR. ITS 3.1.1 ACTION A provides actions when the SDM is not within limits. ITS SR 3.1.1.1 requires verification that the SDM is within limits. This changes the CTS by moving the SDM limits to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications to the COLR is acceptable because the cycle-specific limits are developed or utilized under NRC-approved methodologies that will ensure that the safety limits are met. The NRC documented in Generic Letter 88-16,

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

"Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specification to provide adequate protection of public health and safety. The ITS retains the SDM requirement. The methodologies used to develop the parameters in the COLR have obtained approval by the NRC in accordance with Generic Letter 88-16. Furthermore, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "Core Operating Limits Report." ITS 5.6.3 ensures the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.1.1.1.1.e and CTS 4.1.1.2.b require SDM to be determined to be within its limits every 24 hours. ITS SR 3.1.1.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.1.e and CTS 4.1.1.2.b require determination that the SDM is within limits, and specifically requires the consideration of the following factors: reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration and samarium concentration. ITS SR 3.1.1.1 requires a determination that the SDM is within limits, but does not describe the factors that must be considered in the calculation. This information is moved to the Bases. This changes the CTS by removing details on how the SDM calculation is performed from the Specification and placing the information in the Bases.

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DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement that the SDM be within limits. The detail of how SDM is calculated does not need to appear in the specification in order for the requirement to apply. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.1.1 ACTION states when the SDM is less than the applicable limit, boration must be initiated immediately. ITS 3.1.1 ACTION states when SDM is not within limits, boration must be initiated within 15 minutes. This changes the CTS by relaxing the Completion Time from "immediately" to 15 minutes.

The purpose of CTS 3.1.1.1 ACTION is to restore the SDM to within its limit promptly. This change is acceptable because the Completion Time is consistent with safe operation under the specific Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. This ITS Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. In addition, the ITS Bases for the ACTION states that boration must be initiated promptly. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 ACTION states when the SDM is less than or equal to 1.6% $\Delta k/k$, boration must be initiated and continued at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SDM is restored. ITS 3.1.1 ACTION A states that when the SDM is not within limits to initiate boration to restore SDM to within limits. This changes the CTS by eliminating the specific values of flow rate and the boron concentration used to restore compliance with the LCO.

The purpose of CTS 3.1.1.1 ACTION is to restore the SDM to within its limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

specified redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. Removing the specific values of flow rate and boron concentration from the CTS ACTION provides flexibility in the restoration of the SDM and eliminates conflicts between the SDM value and the specific boration values in the CTS ACTION. As stated, in the ITS Bases for ACTION A, "In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions." Specifying a minimum flow rate and concentration in the ACTION may not accomplish the objective of raising the RCS boron concentration as soon as possible. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.1.1.1.d requires verification that the SDM is within limit, "Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below (CTS 4.1.1.1.1.e), with the control banks at the maximum insertion limit of Specification 3.1.3.6." The ITS does not contain a similar requirement. This changes the CTS by deleting Surveillance Requirement 4.1.1.1.1.d.

The purpose of CTS 4.1.1.1.1.d is to verify core design predictions by determining the SDM with the control rods at the insertion limits. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the LCO is within limit. The core design predictions, such as rod worth, boron worth, and critical boron concentration, are verified in a manner and at a Frequency necessary to give confidence that these predicted values are within limit in accordance with ITS SR 3.1.2.1. ITS SR 3.1.2.1 has a conditional Frequency similar to that of CTS 4.1.1.1.d requiring performance once prior to entering MODE 1 (> 5% RTP) after each refueling. To ensure the SDM is within limits during reactor startup the critical boron concentration is verified during the startup physics test program and prior to criticality per ITS SR 3.1.6.1 (Estimated Critical Position). Thereafter SDM is confirmed by performance of ITS SR 3.1.4.1 (Rod Alignment), SR 3.1.5.1(Shutdown Bank Rod Insertion Limits), and SR 3.1.6.2 (Control Bank Rod Insertion Limits). Thus, the SDM continues to be verified in a manner and at a Frequency necessary to give confidence that the parameter is within limit. Therefore, the core design parameters upon which SDM relies are verified before exceeding 5% RATED THERMAL POWER after each refueling outage. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

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

CTS

SDM
3.1.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

3.1.1.1, 3.1.1.2

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

3.1.1.1 Applicability, 3.1.1.2 Applicability

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.1.1 ACTION, 3.1.1.2 ACTION	A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.1.1.1.1.e, 4.1.1.2.b	SR 3.1.1.1 Verify SDM to be within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 1

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

3.1.1.1, 3.1.1.2

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

3.1.1.1 Applicability, 3.1.1.2 Applicability

APPLICABILITY: MODE 2 with k_{eff} < 1.0, MODES 3, 4, and 5.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.1.1 ACTION, 3.1.1.2 ACTION	A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.1.1.1.e, 4.1.1.2.b	SR 3.1.1.1	Verify SDM to be within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1, SHUTDOWN MARGIN

1. ISTS SR 3.1.1.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a~~n~~ SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis. 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≲ 280 cal/gm energy deposition for the rod ejection accident), and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a ~~guillotine~~ double ended break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1. 1

In addition to the limiting MSLB transient, the SDM requirements must also protect against: 2

a. Inadvertent boron dilution,

b. An uncontrolled rod withdrawal from subcritical or low power condition,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Startup of an inactive reactor coolant pump (RCP), and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or ~~a high pressurizer pressure~~ an overtemperature ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits. 1

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

BASES

LCO (continued)

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid ~~storage~~ tank, or the ~~borated~~ refueling water storage tank. The operator should borate with the best source available for the plant conditions. 1

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k (1000 pcm) must be recovered and a boration flow rate of ~~[ ]~~ 50 gpm, it is possible to increase the boron concentration of the RCS by INSERT 1 ~~100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example~~. 3

1

INSERT 1

147 ppm in approximately 46 minutes. If a boron worth of 6.8 pcm/ppm is assumed, this combination will increase the SDM by 1% $\Delta k/k$ or 1000 pcm. These boration parameters represent Sequoyah typical values and are provided for the purpose of offering a specific example.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with ~~K~~ k $_{eff}$ ≥ 1.0, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth. 2

In MODE 2 with k_{eff} < 1.0 and in MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects: 6

a. RCS boron concentration,

b. Control bank position,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration, and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

[~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
] 4

BASES

REFERENCES	1. 10 CFR 50, Appendix A, GDC 26.
	2. U FSAR, ~~Chapter [15]~~ Section 15.4.2. (1) (3)
	3. U FSAR, ~~Chapter [15]~~ Section 15.2.4. (1) (3)
	4. 10 CFR 100.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a~~n~~ SDM that ensures 2
specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≲ 280 cal/gm energy deposition for the rod ejection accident), and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a ~~guillotine~~ double ended break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1. 1

In addition to the limiting MSLB transient, the SDM requirements must also protect against: 2

a. Inadvertent boron dilution,

b. An uncontrolled rod withdrawal from subcritical or low power condition,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Startup of an inactive reactor coolant pump (RCP), and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or ~~a high pressurizer pressure~~ an overtemperature ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits. 1

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

BASES

LCO (continued)

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid ~~storage~~ tank, or the ~~borated~~ refueling water storage tank. The operator should borate with the best source available for the plant conditions. 1

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k (1000 pcm) must be recovered and a boration flow rate of ~~[~~ 50 ~~]~~ gpm, it is possible to increase the boron concentration of the RCS by INSERT 1 ~~100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example~~. 3

1

INSERT 1

156 ppm in approximately 48 minutes. If a boron worth of 6.4 pcm/ppm is assumed, this combination will increase the SDM by 1% $\Delta k/k$ or 1000 pcm. These boration parameters represent Sequoyah typical values and are provided for the purpose of offering a specific example.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $K_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth. 2

MODE 2 with $k_{eff} < 1.0$ and in

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects: 6

a. RCS boron concentration,

b. Control bank position,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration, and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

[~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~-----~~] 4

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. U FSAR, ~~Chapter [15]~~ Section 15.4.2. 1 3

3. U FSAR, ~~Chapter [15]~~ Section 15.2.4. 1 3

4. 10 CFR 100.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1 BASES, SHUTDOWN MARGIN (SDM)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Editorial changes made for enhanced clarity/consistency.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.1.1.1 Bases provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. Changes are made to be consistent with the Specification.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.1, SHUTDOWN MARGIN

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 2

ITS 3.1.2, CORE REACTIVITY

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

A01

ITS ITS 3.1.2

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

Core Reactivity A02

~~SHUTDOWN MARGIN - T_{avg} Greater Than 200°F~~

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.1.2 A02

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation. See ITS 3.1.1

Applicability APPLICABILITY: MODES 1, 2* ~~, 3, and 4~~. L01

ACTION:

Add proposed ACTIONS A and B L02

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored. See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. See ITS 3.1.4 If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS 3.1.1

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.6

*See Special Test Exception 3.10.1 See ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6. See ITS 3.1.1

SR 3.1.2.1 e. ~~When in MODES 3 or 4~~, Prior to entering MODE 1 after refueling and at least once per 24 hours ~~by consideration of the following factors:~~ L03 LA02

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration.~~

L03

In accordance with the Surveillance Frequency Control Program LA01

SR 3.1.2.1 4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k ~~at least once per 31 Effective Full Power Days (EFPD)~~. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above.~~ LA02

SR 3.1.2.1 Note The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading. L04

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ITS

A01

ITS 3.1.2

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

Core Reactivity

A02

~~SHUTDOWN MARGIN - T_{avg} ≥ 200°F~~

LIMITING CONDITION FOR OPERATION

A02

Add proposed LCO 3.1.2

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

See ITS 3.1.1

Applicability

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

L01

ACTION:

Add proposed ACTIONS A and B

L02

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2, with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

See ITS 3.1.1

* See Special Test Exception 3.10.1

See ITS 3.1.1

A01

ITS

ITS 3.1.2

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 3.1.1

SR 3.1.2.1

e. ~~When in MODES 3 or 4~~, at least once per 24 hours ~~by consideration of the following factors:~~

Prior to entering MODE 1 after refueling and L03

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~ LA02

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration, and~~

~~6. Samarium concentration.~~

L03

In accordance with the Surveillance Frequency Control Program LA01

SR 3.1.2.1

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k ~~at least once per 31 Effective Full Power Days (EFPD)~~. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above.~~ LA02

SR 3.1.2.1 Note

The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading. L04

DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 4.1.1.1.2 requires the overall core reactivity balance to be compared to predicted values to demonstrate agreement within ± 1% Δk/k. However, this Surveillance is currently part of the SHUTDOWN MARGIN Specification. Additionally, CTS 3.1.1.1 is titled SHUTDOWN MARGIN – T_{avg} Greater Than 200°F. A new LCO, ITS LCO 3.1.2, requires the measured core reactivity to be within ± 1% Δk/k of predicted values. Furthermore, ITS 3.1.2 is titled Core Reactivity. This changes the CTS by having a separate Specification for the Core Reactivity requirement and changing the title.

This change is acceptable because the requirements have not changed. Converting the requirement from a Surveillance in the SHUTDOWN MARGIN specification to an LCO is consistent with the ITS format and content guidance. Any technical changes resulting from this change are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

4.1.1.1.2

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS requires the measured core reactivity to be determined to be within ± 1% Δk/k of the predicted value at least every 31 Effective Full Power Days (EFPD). ITS SR 3.1.2.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program. SII

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DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.2 requires comparison of the actual and predicted core reactivity balance and specifically requires consideration of at least those factors stated in Specification 4.1.1.1.1.e. CTS 4.1.1.1.1.e requires determination of SDM and requires the consideration of the following factors: reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. ITS SR 3.1.2.1 requires comparison of the actual and predicted core reactivity, but does not describe the factors that must be considered in the calculation. This information is relocated to the Bases. This changes the CTS by removing details on how the core reactivity balance comparison calculation is performed from the CTS and placing the information in the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This ITS still retains the requirement that the core reactivity balance comparison be within ± 1% $\Delta k/k$. The details of how this comparison is calculated do not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 2 – Relaxation of Applicability)* CTS 4.1.1.1.2 is applicable in MODES 1, 2, 3, and 4. ITS 3.1.2 is applicable in MODES 1 and 2. This changes the CTS

DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

by reducing the applicable MODES in which the core reactivity requirement must be met.

The purpose of CTS Surveillance 4.1.1.1.2 is to verify the core design by comparing the actual and predicted core reactivity. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analysis and licensing basis. The core reactivity balance can only be determined when the reactor is critical (MODES 1 and 2). Additionally, after performing the Surveillance once after each refueling and after 60 EFPD, the Surveillance Frequency is once per 31 EFPD, which continues to accrue when the reactor is critical. Therefore, reducing the applicable MODES from MODES 1, 2, 3, and 4 to MODES 1 and 2 does not result in a reduction of the verification of this important measure of core design accuracy. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 does not contain ACTIONS to follow if the core reactivity balance Surveillance is not met. If the core reactivity balance Surveillance is not met, CTS LCO 3.0.3 would be entered. CTS LCO 3.0.3 requires the plant to be in MODE 3 within 7 hours, MODE 4 within 13 hours, and MODE 5 within 37 hours. ITS 3.1.2 contains ACTIONS to follow if the core reactivity LCO is not met. If the LCO is not met, 7 days are provided to re-evaluate the core design and safety analysis, to determine that the reactor core is acceptable for continued operation, and to establish appropriate operating restrictions and SRs. If these actions are not completed within the 7 days, the plant must be placed in MODE 3 within 6 hours. This changes the CTS by providing 7 days to evaluate and provide compensatory measures for not meeting the core reactivity balance requirement and then requiring entry into MODE 3 instead of requiring an immediate shutdown and entry into MODE 5.

The purpose of CTS 4.1.1.1.2 is to verify the accuracy of the core design by comparing the predicted and actual core reactivity throughout core life. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Should the core reactivity balance requirement not be met, time is required to determine the cause of the disagreement and what adjustments may be needed to the operating conditions of the core. The startup physics testing program is used to verify most of the critical core design parameters, such as control rods worth, boron worth, and moderator temperature coefficient. In addition, there is considerable conservatism in the application of these values in the accident analyses. Therefore, allowing a time to evaluate the difference and make any adjustments to the operational controls is acceptable. The 7 day Completion time is reasonable considering the complexity of the evaluations and the time to meet administrative requirements, such as 10 CFR 50.59 safety evaluation

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DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

preparation and approval. If it cannot be determined within 7 days that the core is acceptable for continued operation, the unit must be shutdown. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.1.1.1.2 requires comparison of the actual and predicted core reactivity balance at least once per 31 Effective Full Power Days (EFPD) and specifically requires consideration of at least those factors stated in Specification 4.1.1.1.1.e. CTS 4.1.1.1.2 also requires the predicted reactivity values to be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. CTS 4.1.1.1.1.e requires the determination of SDM by considering the reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration in MODE 3 or 4. ITS SR 3.1.2.1 requires verifying the measured core reactivity is within $\pm 1\ \%\ \Delta\ k/k$ of the predicted core reactivity values once prior to entering MODE 1 after each refueling and every 31 EFPD thereafter after 60 EFPD. This changes the CTS by not requiring the periodic, at-power core reactivity comparison until core burnup reaches 60 EFPD. Additionally, it allows the initial verification to be performed in MODE 2.

The purpose of CTS 4.1.1.1.2 is to verify the agreement between the actual and predicted core reactivity. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure it provides an acceptable level of equipment reliability. The CTS and ITS require the predicted core reactivity values to be normalized to the actual values prior to exceeding 60 EFPD of core burnup. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after fuel loading, is acceptable, based on the slow rate of core reactivity changes resulting from fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. In addition, CTS 4.1.1.1.1.e Frequency has been changed to ensure core reactivity is within limits prior to entering MODE 1 after each refueling. This change has been designated as less restrictive because Surveillances will be performed less frequently and in different MODES of operation under the ITS than under the CTS.

L04 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.1.1.1.2 requires, in part, that the predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days (EFPD) after each fuel loading. ITS SR 3.1.2.1 contains an SR Note that states the adjustment "may" be performed prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. This changes the CTS by stating that the normalization may be performed prior to 60 EFPD after each fuel loading.

The purpose of adjusting the predicted reactivity values to the core conditions is to allow benchmarking of the design calculations. Making this adjustment 60 EFPD of operation allows sufficient time for the core conditions to reach

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DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

steady state. This change is acceptable because the expectation is to perform the adjusting of the predicted reactivity values to the core conditions. ITS SR 3.1.2.1 still allows the adjustment to take place prior to the 60 EFPD after each fuel loading. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

DOC A02

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

Applicability

APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02	A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
		<u>AND</u>	
		A.2 Establish appropriate operating restrictions and SRs.	7 days
DOC L02	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.1.1.1.1.e, 4.1.1.1.2 SR 3.1.2.1	--------------------------NOTE-------------------------------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. -- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND --------NOTE-------- Only required after 60 EFPD ----------------------- ~~[31 EFPD thereafter~~ ~~OR~~ 2 In accordance with the Surveillance Frequency Control Program ~~]~~ 2

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

DOC A02

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

Applicability

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02 A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	7 days
DOC L02 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.1.1.1.e, 4.1.1.1.2	SR 3.1.2.1	---------------------------NOTE--------------------------------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. --- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND --------NOTE-------- Only required after 60 EFPD ----------------------- ~~[31 EFPD thereafter~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2, CORE REACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS SR 3.1.2.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with ~~other~~ specific variables ~~fixed~~ (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate. 5

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle ~~(BOC)~~ life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at ~~BOC~~ BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond ~~BOC~~ BOL, or that an unexpected change in core conditions has occurred. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at ~~BOC~~ BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. 1

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A~~n~~ SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). 5

BASES

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at ~~BOC~~ BOL. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value ~~must~~ may take place, if required, within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. ~~[~~ The required subsequent Frequency ~~of 31 EFPD~~, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. 1 6 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 3 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. UFSAR, Chapter ~~[~~15~~]~~. 1 4

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with ~~other~~ specific variables ~~fixed~~ (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate. 5

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle ~~(BOC)~~ life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at ~~BOC~~ BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond ~~BOC~~ BOL, or that an unexpected change in core conditions has occurred. 1 1 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at ~~BOC~~ BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. 1

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A~~n~~ SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). 5

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at ~~BOC~~ BOL. The SR is modified by a Note. The 1
Note indicates that the normalization of predicted core reactivity to the measured value ~~must~~ may take place, if required, within the first 60 effective full power 6
days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. ~~[~~ The required subsequent Frequency ~~of 31 EFPD~~, following 2
the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~--]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. UFSAR, Chapter ~~[~~15~~]~~. 1 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2 BASES, CORE REACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS SR 3.1.2.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. Editorial changes made for enhanced clarity/consistency.

6. Changes are made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.2, CORE REACTIVITY

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 3

ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

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

ITS A01 ITS 3.1.3

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3 3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

Applicability APPLICABILITY: Beginning of cycle life (BOL) limit - MODES 1 and 2* ~~only#~~ A02
End of life cycle (EOL) limit - MODES 1, 2 and 3 ~~only#~~

ACTION:

ACTION A, ACTION B a. With the MTC more positive than the BOL limit specified in the COLR operation in MODES 1 and 2 may proceed provided:

ACTION A 1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours
ACTION B or be in ~~HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 within the next 6 hours. ~~These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.~~ A03 A04

2. ~~The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.~~ L01

ACTION C b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

Applicability *With K_{eff} greater than or equal to 1.0

~~#See Special Test Exception 3.10.3~~ A02

ITS

A01

ITS 3.1.3

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1, SR 3.1.3.2

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

SR 3.1.3.1

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

SR 3.1.3.2, SR 3.1.3.2 Notes 1 and 2

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 PPM surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates that MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured and compared to the EOL MTC limit specified in the COLR at least once per 14 EFPD during the remainder of the fuel cycle.

Add proposed SR 3.1.3.2 Note 3

L02

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3 3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

Applicability APPLICABILITY: Beginning of Cycle life (BOL) Limit - Modes 1 and 2* ~~only#~~ A02
End of Cycle Life (EOL) Limit - Modes 1, 2, and 3 ~~only#~~

ACTION:

ACTION A, ACTION B a. With the MTC more positive than the BOL limit specified in the COLR operation in Modes 1 and 2 may proceed provided:

ACTION A 1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or ~~be in HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 A03 within the next 6 hours. ~~These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.~~ A04
ACTION B

2. ~~The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.~~ L01

ACTION C b. With the MTC more negative than the EOL limit specified in the COLR be in HOT SHUTDOWN within 12 hours.

Applicability * With k_{eff} greater than or equal to 1.0

~~# See Special Test Exception 3.10.3~~ A02

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REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1, SR 3.1.3.2

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

SR 3.1.3.1

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

SR 3.1.3.2, SR 3.1.3.2 Notes 1 and 2

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 PPM surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than 300 PPM surveillance limit specified in the COLR, the MTC shall be remeasured and compared to the EOL MTC limit specified in the COLR at least once per 14 EFPD during the remainder of the fuel cycle.

Add proposed SR 3.1.3.2 Note 3 L02

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DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 The Applicability of CTS 3.1.1.3 is modified by footnote # stating "See Special Test Exception 3.10.3." ITS 3.1.3 Applicability does not contain the footnote or a reference to the Special Test Exception. This changes the CTS by not including footnote # in the ITS.

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A03 CTS 3.1.1.3 ACTION a.1 states that if the MTC is more positive than the BOL limit, control rod withdrawal limits must be imposed within 24 hours or the unit must be in HOT STANDBY within the next 6 hours. ITS 3.1.3 ACTION A states that with the MTC not within the BOL limit, establish administrative control rod withdrawal limits within 24 hours or ACTION B requires the unit to be in MODE 2 with k_{eff} < 1.0 within the next 6 hours. This changes the CTS by requiring the unit to be in MODE 2 with k_{eff} < 1.0 instead of HOT STANDBY (i.e., MODE 3).

This change is acceptable because the requirements have not changed. In accordance with CTS LCO 3.0.1, ACTIONS are only required to be followed while in the MODE of Applicability. The CTS BOL MTC limit is only applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. Therefore, under the CTS, the unit does not have to enter MODE 3 because the applicability of the ACTION ends when in MODE 2 with k_{eff} < 1.0. As a result, there is no difference between the CTS and ITS requirements. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.1.1.3 ACTION a.1 states that if the MTC is more positive than the BOL limit, then control rod withdrawal limits must be established. It also states that these withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6. ITS 3.1.3 does not contain this statement. This changes the CTS by not including the statement that the withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

This change is acceptable because the requirements have not changed. The CTS reference to Specification 3.1.3.6 is an "information only" statement that neither adds, eliminates, or modifies requirements. The ITS convention is to not

DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

include these types of statements. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.3 ACTION a.2 states that if the measured MTC is more positive than the BOL limit, then the control rod withdrawal limits established in ACTION a.1 must be maintained until subsequent calculation verifies that the MTC has been restored to within limits for all the rods withdrawn condition. ITS 3.1.3 does not contain a requirement that the control rod withdrawal limits must be maintained until MTC is confirmed to be within its limit by measurement. However, ITS LCO 3.0.2 states that the Required Actions shall be followed until the LCO is met or no longer applicable. The ITS 3.1.3 Bases state that physics calculations may be used to determine the time in cycle life at which the calculated MTC will meet the LCO requirement, and at this point in core life the condition may be exited and the control rod withdrawal limits removed. This changes the CTS by eliminating the requirement to verify the MTC to be within its limit before removing the control rod withdrawal limits.

The purpose of CTS 3.1.1.3 ACTION a.2 is to ensure that the additional operational restrictions required to maintain the MTC within the assumptions in the safety analyses are maintained until the MTC value without the restrictions is within the LCO limits. This change is acceptable because the deleted Action is not necessary to verify that the values used to meet the LCO are consistent with the safety analyses. Thus, appropriate values continue to be tested in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. The measurement of the MTC, boron endpoint, and control rod worth prior to entering MODE 1 is sufficient to verify, the nuclear design so that it can be accurately predicted when the all rods out, full power equilibrium MTC is within the LCO limit. Performing another measurement of beginning of cycle MTC to confirm this prediction is not necessary to give confidence that MTC is within its limit. This change is designated as less restrictive because Actions that are required in the CTS will not be required in the ITS.

DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.1.1.3.b requires MTC to be determined within limits. MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm Surveillance limit specified in the COLR. In the event this comparison indicates that the MTC is more negative than 300 PPM surveillance limit specified in the COLR, MTC shall be remeasured and compared to the EOL MTC limit specified in the COLR at least once per 14 EFPD during the remainder of the fuel cycle. ITS SR 3.1.3.2 requires verifying MTC is within the EOL limit once each cycle. Additionally, ITS SR 3.1.3.2 is modified by three notes. The first Note states that ITS SR 3.1.3.2 is not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. The second Note states that if the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, then ITS SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. The third Note states that ITS SR 3.1.3.2 does not need to be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. This changes the CTS by eliminating the requirement to verify that MTC is met at least once per 14 EFPD if the measured MTC at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR.

The purpose of CTS 4.1.1.3.b is to periodically verify that the MTC EOL limit is within limit if the 300 ppm Surveillance limit in the COLR is not met. This change is acceptable because the Surveillance Frequency has been evaluated to ensure it will provide an acceptable level of assurance that the MTC EOL limit is not exceeded. This will help ensure that the MTC EOL limit is not exceeded for the remainder of the cycle. The new 60 ppm Surveillance limit will be incorporated into the COLR. This new limit is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, then the MTC EOL limit will not be exceeded because the gradual manner in which MTC changes with core burnup. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

3.1.1.3 LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ~~[≤ [ ]~~ < 0 Δk/k°F ~~at hot zero power] [that specified in Figure 3.1.3-1]~~. 1

Applicability APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the ~~upper~~ beginning of cycle life (BOL) MTC limit, MODES 1, 2, and 3 for the ~~lower~~ end of cycle life (EOL) MTC limit. 2

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a.1	A. MTC not within ~~upper~~ BOL limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours	2
ACTION a.1	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours	
ACTION b	C. MTC not within ~~lower~~ EOL limit.	C.1 Be in MODE 4.	12 hours	2

SURVEILLANCE REQUIREMENTS

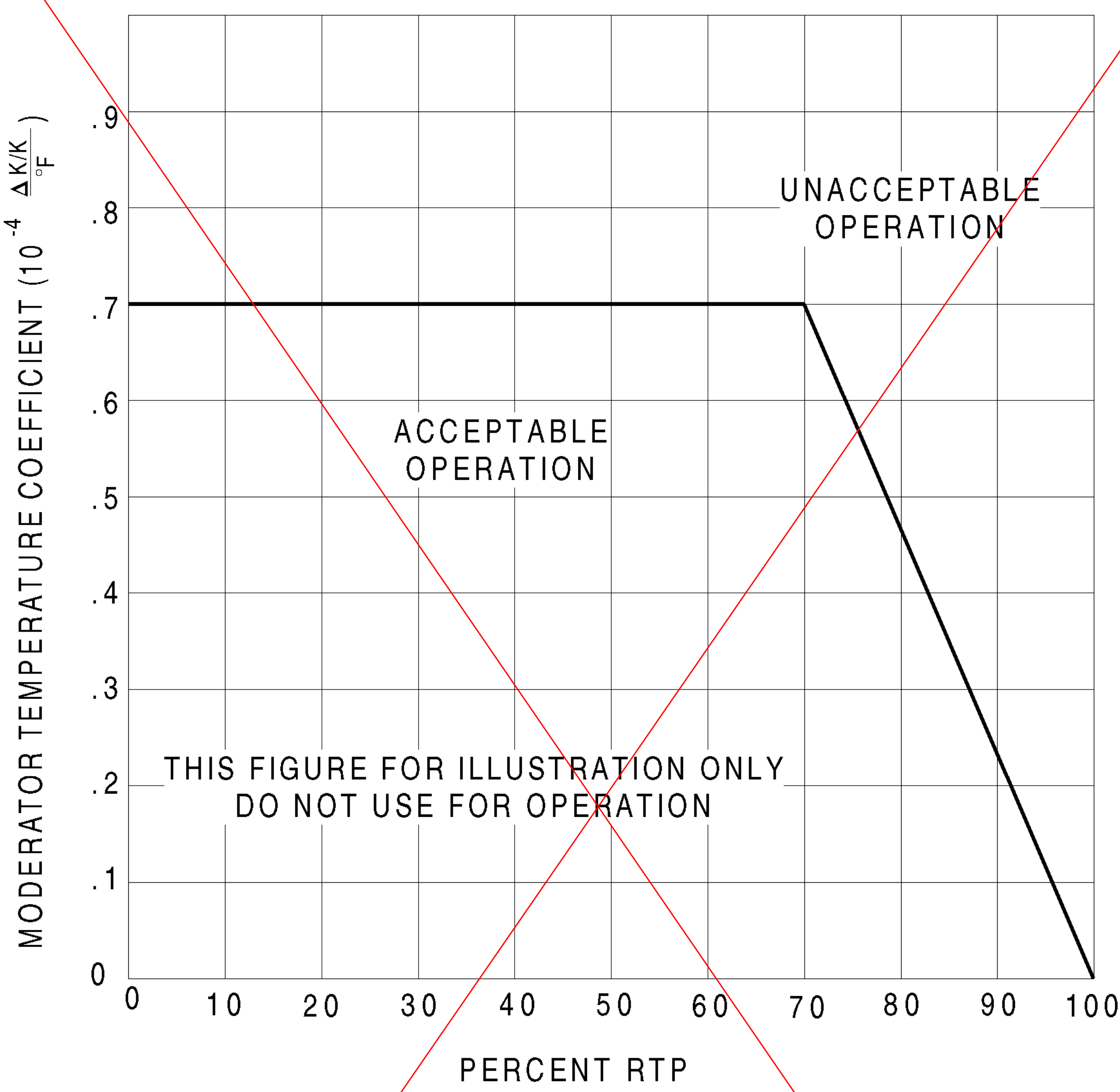
	SURVEILLANCE		FREQUENCY	
4.1.1.3.a	SR 3.1.3.1	Verify MTC is within ~~upper~~ BOL limit.	Prior to entering MODE 1 after each refueling	2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.1.1.3.b	SR 3.1.3.2	---------------------------NOTES------------------------------- 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. 2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. --- Verify MTC is within ~~lower~~ EOL limit.	Once each cycle 2

CTS

MTC
3.1.3



3

Figure 3.1.3 - 1 (page 1 of 1)
Moderator Temperature Coefficient Vs. Rated Thermal Power

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CTS

MTC
3.1.3

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

3.1.1.3

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ~~[≤ [ ]~~ < 0 Δk/k°F ~~at hot zero power] [that specified in Figure 3.1.3-1]~~. 1

Applicability

APPLICABILITY: MODE 1 and MODE 2 with k_{eff} ≥ 1.0 for the ~~upper~~ beginning of cycle life (BOL) MTC limit,
MODES 1, 2, and 3 for the ~~lower~~ end of cycle life (EOL) MTC limit. 2

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a.1	A. MTC not within ~~upper~~ BOL limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours	2
ACTION a.1	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with k_{eff} < 1.0.	6 hours	
ACTION b	C. MTC not within ~~lower~~ EOL limit.	C.1 Be in MODE 4.	12 hours	2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.1.1.3.a	SR 3.1.3.1 Verify MTC is within ~~upper~~ BOL limit.	Prior to entering MODE 1 after each refueling	2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
4.1.1.3.b SR 3.1.3.2	---------------------------NOTES-------------------------------- 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. 2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. -- Verify MTC is within ~~lower~~ EOL limit.	Once each cycle	2

CTS

MTC
3.1.3

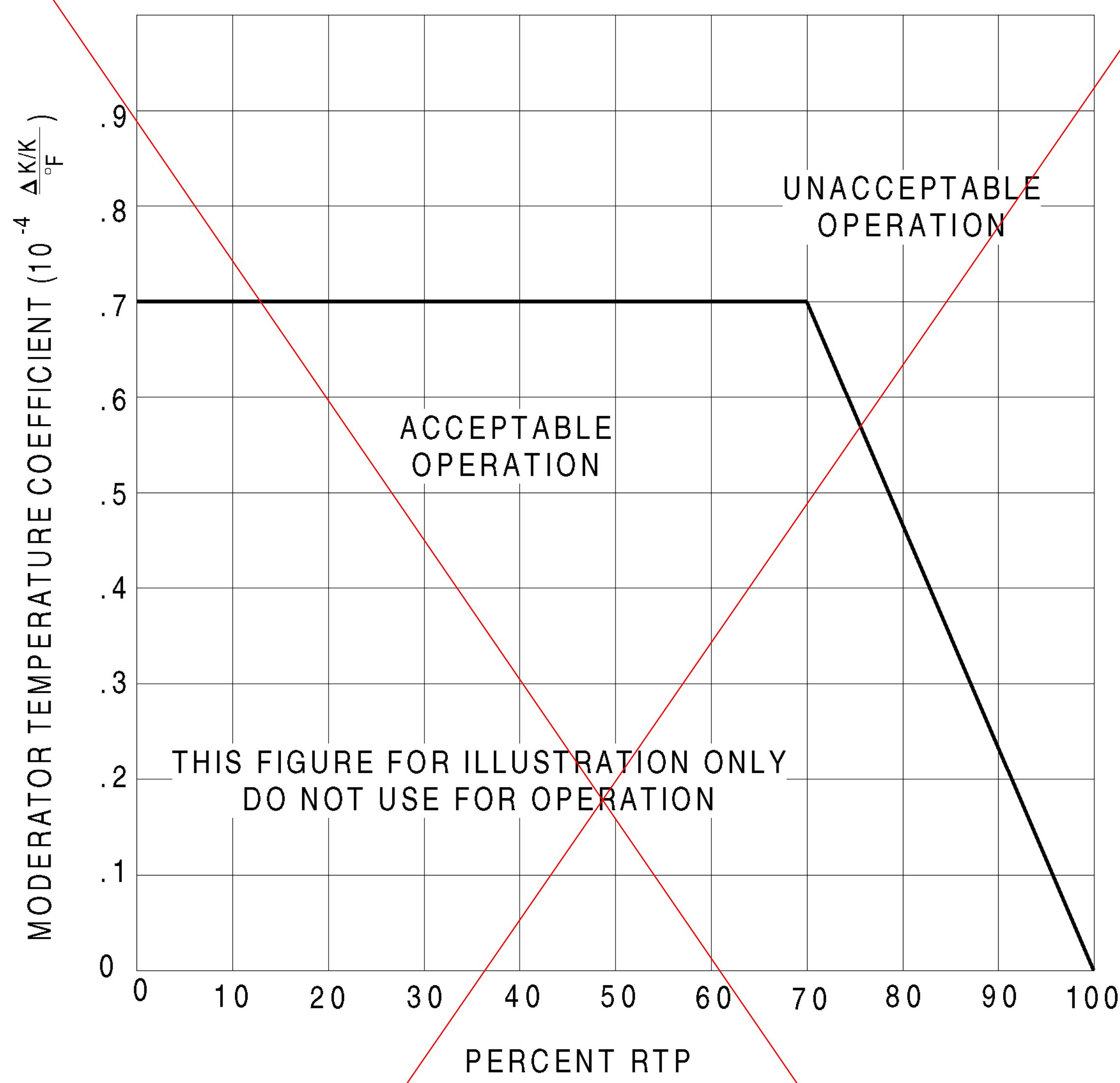

Figure 3.1.3 - 1 (page 1 of 1)
Moderator Temperature Coefficient Vs. Rated Thermal Power

3

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS 3.1.3 contains Figure 3.1.3-1 for Moderator Temperature Coefficient Vs Rated Thermal Power. This figure is not maintained in ITS 3.1.3. ITS 3.1.3 lists the maximum upper limit value in the LCO. Therefore, ISTS Figure 3.1.3-1 is not required and has been deleted.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) ~~(BOC)~~ MTC is less than zero when THERMAL 1
POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL ~~BOC~~ within the range analyzed in the plant 1
accident analysis. The end of cycle life (EOL) ~~(EOC)~~ MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL ~~EOC~~ limit. 1

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR 1
accident and transient analyses.

BASES

BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The U FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3). 1

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the ~~BOC or EOC life~~ BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2). 1

MTC values are bounded in reload safety evaluations assuming steady state conditions at ~~BOC and EOC~~ BOL and EOL. An ~~EOC~~ EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the ~~EOC~~ EOL value, in order to confirm reload design predictions. 1

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be maintained within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. 4

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at ~~BOC~~ BOL; this upper bound must not be exceeded. This maximum upper limit occurs at ~~BOC~~ BOL, all rods out (ARO), hot zero power conditions. At ~~EOC~~ EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded. 1

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at ~~BOC~~ BOL and ~~EOC~~ EOL on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met. 1

BASES

LCO (continued)

The LCO establishes a maximum positive value that cannot be exceeded. The ~~BOC~~ BOL positive limit and the ~~EOC~~ EOL negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule. 1

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the ~~upper~~ BOL limit must ~~also~~ be maintained to ensure that startup and 5 3
subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The ~~lower~~ EOL MTC limit must be maintained in MODES 2 and 3, in 5
addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the ~~BOC~~ BOL MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits. 1

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

BASES

ACTIONS (continued)

B.1

If the required administrative withdrawal limits at ~~BOC~~ BOL are not established within 24 hours, the unit must be brought to at least MODE 2 with k_{eff} < 1.0 to prevent operation with an MTC that is more positive than that assumed in safety analyses. 1 3

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the ~~EOC~~ EOL MTC limit means that the safety analysis assumptions for the ~~EOC~~ EOL accidents that use a bounding negative MTC value may be invalid. If the ~~EOC~~ EOL MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours. 1

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at ~~BOC~~ BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels. 1

The ~~BOC~~ BOL MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the ~~BOC~~ BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for ~~EOC~~ EOL full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the ~~EOC~~ EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the ~~EOC~~ EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met. 1 1

SR 3.1.3.2 is modified by three Notes that include the following requirements:

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the ~~EOC~~ EOL limit on MTC could be reached before the planned ~~EOC~~ EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the ~~EOC~~ EOL limit. 1

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the ~~EOC~~ EOL limit will not be exceeded because of the gradual manner in which MTC changes with core burnup. 1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. U FSAR, Chapter ~~[~~15~~]~~. 1 2
3. ~~WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985~~ BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989. 1
4. U FSAR, Chapter ~~[~~15 .2.1~~]~~. 1 2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) ~~(BOC)~~ MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL ~~BOC~~ within the range analyzed in the plant accident analysis. The end of cycle life (EOL) ~~(EOC)~~ MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL ~~EOC~~ limit. 1

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses. 1

BASES

BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

U

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3). 1

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the ~~BOC or EOC life~~ BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2). 1

MTC values are bounded in reload safety evaluations assuming steady state conditions at ~~BOC and EOC~~ BOL and EOL. An ~~EOC~~ EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the ~~EOC~~ EOL value, in order to confirm reload design predictions. 1

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be maintained within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. 4

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at ~~BOC~~ BOL; this upper bound must not be exceeded. This maximum upper limit occurs at ~~BOC~~ BOL, all rods out (ARO), hot zero power conditions. At ~~EOC~~ EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded. 1

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at ~~BOC~~ BOL and ~~EOC~~ EOL on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met. 1

BASES

LCO (continued)

The LCO establishes a maximum positive value that cannot be exceeded. The BOL ~~BOC~~ positive limit and the EOL ~~EOC~~ negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule. 1

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the BOL ~~upper~~ limit must ~~also~~ be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The EOL ~~lower~~ MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES. 5 3 5

ACTIONS

A.1

If the BOL ~~BOC~~ MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits. 1

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

BASES

ACTIONS (continued)

B.1

If the required administrative withdrawal limits at ~~BOC~~ BOL are not established within 24 hours, the unit must be brought to at least MODE 2 with k_{eff} < 1.0 to prevent operation with an MTC that is more positive than that assumed in safety analyses. 1 3

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the ~~EOC~~ EOL MTC limit means that the safety analysis assumptions for the ~~EOC~~ EOL accidents that use a bounding negative MTC value may be invalid. If the ~~EOC~~ EOL MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours. 1

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at ~~BOC~~ BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels. 1

The ~~BOC~~ BOL MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the ~~BOC~~ BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

EOL

In similar fashion, the LCO demands that the MTC be less negative than the specified value for ~~EOC~~ full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the ~~EOC~~ EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the ~~EOC~~ EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met. 1 1

SR 3.1.3.2 is modified by three Notes that include the following requirements:

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the ~~EOC~~ EOL limit on MTC could be reached before the planned ~~EOC~~ EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the ~~EOC~~ EOL limit. 1

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the ~~EOC~~ EOL limit will not be exceeded because of the gradual manner in which MTC changes with core burnup. 1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. U FSAR, Chapter ~~[~~15~~]~~. 1 2

3. ~~WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985~~. BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989 1

4. U FSAR, Chapter ~~[~~15.2.1~~]~~. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3 BASES, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Editorial changes made for enhanced clarity/consistency.

4. Changes are made to be consistent with the Specification.

5. Changes are made to be consistent with changes made to the Specification.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 4

ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

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

ITS A01 ITS 3.1.4

REACTIVITY CONTROL SYSTEMS

~~3/4.1.3 MOVABLE CONTROL ASSEMBLIES~~ Alignment Limits A01

Rod GROUP ~~HEIGHT~~

LIMITING CONDITION FOR OPERATION

LCO 3.1.4 3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

Applicability APPLICABILITY: MODES 1~~*~~ and 2~~*~~ A02

ACTION:

ACTION A a. With one or more full length rods untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours. Add proposed Required Action A.1.2 L01

ACTION D b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours. Add proposed Required Action D.1.1 and D.1.2 M01

ACTION B c. With one full length rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored within the above alignment requirements, or

2. ~~The remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limit of specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or~~ A03

3. ~~The rod is declared inoperable and~~ the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that: L02 Add proposed Required Action B.2.1.2 L01

~~*See Special Test Exceptions 3.10.2 and 3.10.3.~~ A02

REACTIVITY CONTROL SYSTEMS

ACTION: (Continued)

ACTION B

a) A reevaluation of ~~each~~ accident analysis ~~of Table 3.1-1~~ is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions. LA03

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within ~~one~~ two hour ~~and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER~~. L03 L04

Add proposed ACTION C M02

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions ~~at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours~~. In accordance with the Surveillance Frequency Control Program LA01 L05

SR 3.1.4.2 4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be trippable by verifying rod freedom of movement by movement of ≥ 10 steps in either direction ~~at least once per 92 days~~. In accordance with the Surveillance Frequency Control Program LA01

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ITS

A01

ITS 3.1.4

~~TABLE 3.1-1~~

~~ACCIDENT ANALYSES REQUIRING REEVALUATION~~
~~IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD~~

~~Rod Cluster Control Assembly Insertion Characteristics~~

~~Rod Cluster Control Assembly Misalignment~~

~~Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System~~

~~Single Rod Cluster Control Assembly Withdrawal At Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)~~

~~Major Secondary System Pipe Rupture~~

~~Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)~~

LA03

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REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

SR 3.1.4.3 3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position~~#~~ shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: LA02

a. T_{avg} greater than or equal to ~~541~~ 500°F, and L06

b. All reactor coolant pumps operating.

Applicability APPLICABILITY: MODES 1 and 2

ACTION:

~~a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ A04

~~b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER~~ M03

Add proposed ACTION A M04

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

~~b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and~~ L07

~~c. At least once per 18 months.~~ L08

~~#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.~~ LA02

ITS

A01

ITS 3.1.4

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} Greater Than 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

~~4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:~~

L09

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

See ITS 3.1.1

*See Special Test Exception 3.10.1

See ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k. See ITS 3.1.1

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

~~4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:~~ L09

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS Chapter 1.0

b. At least once per 24 hours by consideration of the following factors:

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

See ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

~~3/4.1.3 MOVABLE CONTROL ASSEMBLIES~~

Rod GROUP HEIGHT Alignment Limits A01

LIMITING CONDITION FOR OPERATION

LCO 3.1.4 3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

Applicability APPLICABILITY: Modes 1~~*~~ and 2~~*~~. A02

ACTION:

ACTION A a. With one or more full length rods untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour Add proposed Required Action A.1.2 and be in HOT STANDBY within 6 hours. L01

ACTION D b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), Add proposed Required Action D.1.1 and D.1.2 be in HOT STANDBY within 6 hours. M01

ACTION B c. With one full length rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored within the above alignment requirements, or

2. ~~The remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limit of specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or~~ A03

3. ~~The rod is declared inoperable and~~ the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. Add proposed Required Action B.2.1.2 POWER OPERATION may then continue provided that: L02 L01

   a) A reevaluation of ~~each~~ accident analysis ~~of Table 3.1-1~~ is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions. LA03

~~* See Special Test Exceptions 3.10.2 and 3.10.3.~~ A02

REACTIVITY CONTROL SYSTEMS

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

ACTION B

c) A power distribution map is obtained from the movable incore detectors $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within ~~one~~ two hour ~~and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER~~.

L03

L04

Add proposed ACTION C

M02

SURVEILLANCE REQUIREMENTS

In accordance with the Surveillance Frequency Control Program

LA01

SR 3.1.4.1

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions ~~at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours~~.

L05

SR 3.1.4.2

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be trippable by verifying rod freedom of movement by movement of ≥ 10 steps in either direction ~~at least once per 92 days~~.

In accordance with the Surveillance Frequency Control Program

LA01

ITS

A01

ITS 3.1.4

~~TABLE 3.1-1~~

~~ACCIDENT ANALYSES REQUIRING REEVALUATION~~
~~IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD~~

~~Rod Cluster Control Assembly Insertion Characteristics~~

~~Rod Cluster Control Assembly Misalignment~~

~~Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System~~

~~Single Rod Cluster Control Assembly Withdrawal At Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)~~

~~Major Secondary System Pipe Rupture~~

~~Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)~~

LA03

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REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

LA02

SR 3.1.4.3 3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position ~~#~~ shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to ~~541~~ 500 °F, and L06

b. All reactor coolant pumps operating.

Applicability APPLICABILITY: Modes 1 and 2.

ACTION:

~~a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ A04

~~b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER.~~ M03

Add proposed ACTION A M04

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

~~b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and~~ L07

~~c. At least once per 18 months.~~ L08

~~# Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥222 and ≤231 steps withdrawn, inclusive.~~ LA02

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \geq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation. See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

~~4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:~~ L09

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2, with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.1

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

* See Special Test Exception 3.10.1 See ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k. See ITS 3.1.1

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

~~4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:~~ L09

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS Chapter 1.0

b. At least once per 24 hours by consideration of the following factors: See ITS 3.1.1

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

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DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.1.3.1 Applicability is modified by Footnote * which states "See Special Test Exceptions 3.10.2 and 3.10.3." ITS 3.1.4 Applicability does not contain this Note. This changes the CTS by not including Footnote *.

The purpose of Footnote * is to alert the Technical Specification user that a Special Test Exception exists that may modify the Applicability of this Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.1.3.1 ACTION c.2 states that with one full length rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour, the remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limit of specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation. ITS 3.1.4 does not contain a Required Action stating that the remainder of the rods in the group must be aligned with the misaligned rod. This changes the CTS by not including a specific Required Action stating that the remainder of the rods in the group must be aligned with the misaligned rod.

This change is acceptable because the technical requirements have not changed. The moving of the remaining rods to within the LCO limit of the misaligned rod, while complying with all of the other rod position requirements, is simply restoring compliance with the LCO. Restoration of compliance with the LCO is always an available Required Action and it is the convention of the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 3.1.3.4 ACTION a states with the drop time of any full length rod determined to exceed the above limit restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2. ITS 3.1.4 does not have a similar requirement. This changes the CTS by not explicitly requiring, in the ITS 3.1.4 ACTIONS, restoration of the rod drop time prior to proceeding to MODE 1 or 2.

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DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

CTS 4.0.4 and ITS SR 3.0.4 require verification that Surveillances are met prior to entering the MODE in which they apply. CTS 4.0.4 and ITS SR 3.0.4 also prohibit entering a MODE or condition with the Surveillance not met and while relying on actions. Therefore, since the Applicability of CTS 3.1.3.4 is MODES 1 and 2, the action prohibiting entry into MODES 1 and 2 with the rod drop time requirements not met is redundant to CTS 4.0.4 and ITS 3.0.4. This change is acceptable because the technical requirements have not changed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.1 ACTION b states "With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours." ITS 3.1.4 ACTION D adds additional requirements (ITS 3.1.4 Required Actions D.1.1 and D.1.2) to verify SHUTDOWN MARGIN is within the limits within 1 hour or to initiate boration to restore the required SHUTDOWN MARGIN to within limits. This changes the CTS by adding two additional Required Actions.

The purpose of CTS 3.1.3.1 ACTION a is to place the unit in a MODE in which the equipment is not required. More than one control rod misaligned from its group average has the potential to reduce the SHUTDOWN MARGIN. Therefore, the SHUTDOWN MARGIN must be evaluated. ITS 3.1.4 adds Required Actions to allow verification that the SHUTDOWN MARGIN is within the limit or to borate to restore the SHUTDOWN MARGIN to within limits. These new Required Actions must be accomplished within 1 hour. The one hour allows the operator adequate time to determine the SHUTDOWN MARGIN. Restoration of the required SHUTDOWN MARGIN, if necessary, requires increasing the RCS boron concentration to provide negative reactivity. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete this action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SHUTDOWN MARGIN is restored. This change is acceptable because it is consistent with the assumptions of the safety analyses to be within the SHUTDOWN MARGIN limit. This change has been designated as more restrictive because it adds explicit actions to verify SHUTDOWN MARGIN or to restore SHUTDOWN MARGIN within limits.

M02 CTS 3.1.3.1 ACTION c requires that with one full length rod misaligned, POWER OPERATION may continue provided certain actions are completed within one hour. If those actions are not complete, CTS 3.0.3 is required to be entered since no further actions are specified. CTS 3.0.3 allows 1 hour to initiate action and 6 additional hours for the unit to be placed in MODE 3. ITS 3.1.4 ACTION C states that if the Required Action and associated Completion Time of Condition B is not met, the unit must be in MODE 3 within 6 hours. This changes the CTS by providing a specific default condition instead of requiring entry into CTS 3.0.3, and thereby reduces the time to reach MODE 3 following discovery of a misaligned rod if Required Actions are not met from 7 hours to 6 hours.

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DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

The purpose of requiring a shutdown when a rod misalignment cannot be corrected is to bring the unit to a subcritical condition prior to the buildup of an undesirable reactor core power distribution. This change is acceptable because the proposed default condition will require the plant to be in a condition where the rod group alignment limits are no longer applicable. The proposed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power in an orderly manner and without challenging unit systems. This change is designated as more restrictive since the 1 hour specified in CTS 3.0.3 no longer applies.

M03 CTS 3.1.3.4 ACTION b provides an allowance for operation to proceed with THERMAL POWER restricted to less than or equal to 71% of RATED THERMAL POWER, with rod drop times within limits but determined with 3 reactor coolant pumps operating. ITS 3.1.4 does not contain a similar allowance. This changes the CTS by not allowing continued operation at reduce power when the rod drop times are determined with only 3 reactor coolant pumps operating.

The purpose of CTS 3.1.3.4 is to ensure the rods insert within the rod drop criteria. This change is acceptable because ITS SR 3.1.4.3 requires verification of the rod drop times be performed with all of the RCPs operating and the average moderator temperature is ≥ 500°F. Therefore, ITS 3.1.4 will not allow the rod drop times to be determined with only 3 reactor coolant pumps operating. This change is designated as more restrictive because an allowance is being removed from the CTS.

M04 CTS 3.1.3.4 ACTION a requires that with the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2. CTS 3.1.3.4 ACTION b requires that with the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER. However, no specific actions are stated in CTS 3.1.3.4 when the unit is in MODES 1 and 2 when the drop time is discovered to not be within limits. Therefore, CTS 3.0.3 entry would be required. CTS 3.0.3 allows one hour to prepare for a shutdown and requires the unit to be in HOT STANDBY (MODE 3) within 7 hours. ITS 3.1.4 ACTION A applies with one or more rods inoperable. ITS 3.1.4 ACTION A requires verification that the SDM is within the limits specified in the COLR or initiate boration to restore the SDM to within limit within one hour, and to be in MODE 3 within 6 hours. This changes the CTS by adding new requirements associated with SDM and changing the requirement to be outside of the MODE of Applicability from 7 hours to 6 hours.

The purpose of requiring a shutdown when a drop time of any full length rod is not met is to bring the unit to a subcritical condition. With one or more slow control rod(s) there is a potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution in the reactor core, the low probability of an

accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. In addition, the new time to reach MODE 3 is consistent with the time provided in other specifications. This change is acceptable because it is consistent with the requirements of the assumptions of the safety analyses to be within the SDM limit. The change has been designated as more restrictive because it adds explicit actions to verify SDM or to restore SDM within limits and reduces the time required to be in MODE 3.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.1.3.1.1 requires that the position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours. CTS 4.1.3.1.2 requires each full-length rod not fully inserted in the core shall be determined to be trippable by verifying rod freedom of movement by movement of ≥ 10 steps in either direction at least once per 92 days. ITS SR 3.1.4.1 and SR 3.4.1.2 require similar Surveillances and specify the periodic Frequencies as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.4 requires the individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to

DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

dashpot entry with T_{avg} greater than or equal to 541°F and all reactor coolant pumps operating. Additionally, it contains a footnote (footnote #) which states "Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥222 and ≤231 steps withdrawn, inclusive." ITS 3.1.4 does not contain the footnote. This changes the CTS by relocating the footnote to the Bases.

The removal of these details, that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for performing rod drop time testing from the fully withdrawn position. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.1.3.1 ACTION c.3.a) states when a rod is misaligned, POWER OPERATION may continue if a reevaluation of each accident analysis in Table 3.1-1 is performed within 5 days. This reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions. ITS 3.1.4 Required Action B.2.6 states that when one rod is misaligned, re-evaluate the safety analyses and confirm results remain valid for the duration of operation under these conditions. This changes the CTS by moving the accidents listed in Table 3.1-1 to the UFSAR.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to re-evaluate the safety analyses and confirm results remain valid for the duration of operation under these conditions. Additionally, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to procedural detail is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 ACTION a states, in part, with one or more full length rods untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour. CTS 3.1.3.1 ACTION c.3 states, in part, with one full length rod misaligned from its group step counter demand height by more than ± 12 steps (indicated

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position), the rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour. ITS 3.1.4 ACTION A and B requires, within 1 hour, to verify SHUTDOWN MARGIN is within the limits specified in the COLR or to initiate boration to restore SDM to within limits. This changes the CTS by allowing boration to restore SHUTDOWN MARGIN.

The purpose of CTS 3.1.3.1 ACTION a and c.3 is to verify adequate SHUTDOWN MARGIN exists. This change is acceptable because the ITS 3.1.4 Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair the inoperable features. When a rod is inoperable or misaligned, boration may be required to reestablish compliance with the SHUTDOWN MARGIN requirements. Providing a short period of time to reestablish the SHUTDOWN MARGIN requirement instead of entering ITS LCO 3.0.3 is justified because of the existing conservatisms in the SHUTDOWN MARGIN calculations. This change has been designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 ACTION c specifies the requirements for one full length rod misaligned from its group step counter demand height by more than the allowed rod alignment. CTS 3.1.3.1 ACTION c.3 requires the affected rod to be declared inoperable. ITS 3.1.4 ACTION B specifies requirements for one rod not within alignment limits and does not require that the rod be declared inoperable. This changes the CTS by deleting the requirement to declare a misaligned rod inoperable.

The purpose of ITS 3.1.4 is to ensure that the shutdown and control rods are capable of performing their safety function of inserting into the core when required. A secondary function of the control rods is to maintain alignment so that the reactor core power distribution is consistent with the safety analyses. This change is acceptable because the LCO requirements continue to ensure that structures, systems, and components are maintained consistent with the safety analyses and licensing basis. In the ITS, rod OPERABILITY is related only to trippability, and a misaligned rod is not considered inoperable if it can be tripped. Misalignment is addressed by the ITS 3.1.4 LCO, but is separate from OPERABILITY. In both cases, trippability and misalignment, the ITS continues to provide appropriate compensatory measures. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 (*Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 ACTION c.3.d) states that with one rod misaligned, reduce the THERMAL POWER level to less than 75% of the RATED THERMAL POWER within one hour. ITS 3.1.4 Required Action B.2.2 requires THERMAL POWER to be reduced to 75% of the RATED THERMAL POWER within two hours. This changes the CTS by changing the Completion Time from one hour to two hours.

The purpose of CTS 3.1.3.1 ACTION c.3.d) is to reduce reactor core power to ensure that the increases in linear heat generation rate due to misalignment of a

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rod does not result in exceeding the design limits. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Trip System. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L04 (*Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 ACTION c.3.d) states that with one rod misaligned, reduce the high neutron flux setpoint to less than or equal to 85% of RATED THERMAL POWER within the next 4 hours. ITS 3.1.4 Required Action B.2.2 requires THERMAL POWER to be reduced to ≤ 75% RTP, but does not require the high neutron flux trip setpoint to be reduced. This changes the CTS by eliminating the Required Action to reduce the high neutron flux trip setpoint.

The purpose of CTS 3.1.3.1 ACTION c.3.d) is to reduce reactor core power to ensure that the increases in linear heat generation rate due to misalignment of a rod does not result in exceeding the design limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, the capacity and capability of remaining features, and a low probability of a DBA occurring during the repair period. Lowering the high neutron flux trip setpoint increases the chance of an inadvertent reactor trip due to the changes being made to the Reactor Trip System without providing commensurate amount of added safety. Administrative methods of maintaining reactor power below that allowed by the Required Action are sufficient to protect the core. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L05 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.1.3.1.1 states that the position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verifying the group positions at least once per 4 hours. ITS SR 3.1.4.1 requires verifying individual rod positions are within alignment limits in accordance with the Surveillance Frequency Control Program. This changes the CTS by eliminating the requirements to verify the individual rod position to be within alignment limits every 4 hours when the Rod Position Deviation Monitor is inoperable. See DOC LA01 for the relocation of the CTS 4.1.3.1.1 Frequency to the Surveillance Frequency Control Program.

The purpose of CTS 4.1.3.1.1 is to periodically verify that the rods are within the alignment limits specified in the LCO. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of rod position verification when the Rod Position Deviation Monitor is inoperable is unnecessary, since an inoperability of the alarm does not increase the probability

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that the rods are misaligned. The Rod Deviation Monitor, as described in the safety analysis is indication only and is not credited for any automatic action; however, it is there to alert the operator to a dropped rod or misaligned rod by more than 5% span. Its use is not credited in the safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L06 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.1.3.4 requires the individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with T_{avg} greater than or equal to 541°F and all reactor coolant pumps operating. ITS SR 3.1.4.3 specifies the rod drop time be verified at an RCS T_{avg} of ≥ 500°F. This changes the CTS by lowering the required temperature at which rod drop time must be verified.

The purpose of CTS 3.1.3.4 is to ensure the rods insert within the rod drop time criteria. The performance of rod drop time tests ensures that the required negative reactivity insertion (amount and rate) from a reactor trip is within the values assumed in the safety analyses. This change will allow rod drop testing to begin earlier during a startup following a refueling outage. The proposed change is acceptable because the specified rod drop time remains unchanged and the proposed 500°F test temperature is conservative compared to the CTS requirement of 541°F. Since the moderator becomes denser as the RCS temperature is decreased, a lower RCS temperature results in slower rod drops due to the density change of the water. However, the limiting rod drop time requirement of the CTS (2.7 seconds) is maintained in the ITS and must still be met. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L07 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.4.b requires the rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods. ITS 3.1.4 does not contain this testing requirement. This changes the CTS by not explicitly requiring post-maintenance testing on full length rods.

The purpose of CTS 4.1.3.4.b is to verify OPERABILITY of the control rods following maintenance that could alter their operation. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post-maintenance testing is required to demonstrate the OPERABILITY of the system or component. This is described in the Bases for ITS SR 3.0.1 and required under ITS SR 3.0.1. The OPERABILITY requirements for the rod control system are described in the Bases for ITS 3.1.4. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control) provide

DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B, is required under the unit operating license. As a result, post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L08 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.4 requires drop testing of full length rods to be demonstrated through measurement prior to reactor criticality following each removal of the reactor vessel head and at least once per 18 months. ITS 3.1.4.3 requires the test to be performed prior to criticality after each removal of the reactor head. This changes the CTS by deleting the requirement to perform this test at least once per 18 months.

The purpose of CTS 4.1.3.4 is to ensure the rods insert within the rod drop criteria. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its safety function. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence the equipment can perform its assumed safety function. The requirements in the CTS to perform the test following each removal of the reactor vessel head and at least once per 18 months normally coincide with one another. The head is removed once per 18 months unless there is a need to remove the head prior to the end of the cycle. This change is designated as less restrictive because a Surveillance that was required in the CTS will not be performed in the ITS.

L09 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.1.1.1.a requires the SHUTDOWN MARGIN to be determined to be greater than or equal to 1.6% delta k/k within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod is inoperable. CTS 4.1.1.2.a requires the SHUTDOWN MARGIN to be determined to be greater than or equal to 1.0% delta k/k within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod is inoperable. These requirements are applicable in MODES 1, 2, 3, 4, and 5. ITS 3.1.4 Required Action A.1.1 requires the verification of SDM to be within limits within 1 hour. This verification is required in MODES 1 and 2 with one or more control rod(s) inoperable. This changes the CTS by not requiring any explicit SDM verifications for inoperable control rod(s) in MODES 3, 4, and 5 other than the normal verifications specified in ITS SR 3.1.1.1 (once every 24 hours). For MODES 1 and 2 operations, this changes the CTS by not requiring the verification of SDM on a once per 12 hour basis for one or more inoperable rod(s).

The purpose of CTS 4.1.1.1.1.a and CTS 4.1.1.2.a is to provide the appropriate compensatory measures to determine SDM when control rod(s) are inoperable during operations in MODES 1, 2, 3, 4, and 5. The purpose of the ITS 3.1.4 ACTIONS are to provide the appropriate compensatory actions for inoperable control rods in MODES 1 and 2. The purpose of ITS SR 3.1.1.1 is to provide the normal Frequency for verification of SDM regardless of the status of the control rod(s). When the plant is operating in MODES 1 and 2, with one or more rod(s) inoperable, the unit must be in MODE 3 within 6 hours. After reaching MODE 3, ITS 3.1.4 no longer applies therefore it is inappropriate to specify additional

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DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

actions after the unit is outside the Applicability of the Specification. Nevertheless, SDM must still be verified in accordance with ITS SR 3.1.1.1 every 24 hours. This SDM verification must also compensate for the reactivity worth of the control rod that is not fully inserted since it is required by the definition of SDM. Therefore, ITS 3.1.4 ACTIONS provide the appropriate compensatory measures. In MODES 3 and 4, SDM will be monitored in accordance with ITS SR 3.1.1.1 every 24 hours. This change is acceptable since SDM will still be required to be monitored every 24 hours, and based on the definition of SDM the reactivity worth of any rod not capable of being fully inserted must be accounted for in the determination of SDM. Thus, SDM continues to be monitored in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

3.1.3.1

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

3.1.3.1 Applicability, 3.1.3.4 Applicability

APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.3.1 ACTION a, 4.1.1.1.1, 4.1.1.2, DOC M04	A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Be in MODE 3.	1 hour 1 hour 6 hours
3.1.3.1 ACTION c	B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits. OR B.2.1.1 Verify SDM to be within the limits specified in the COLR. OR	1 hour 1 hour

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.1.3.1 ACTION c		B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour	
		AND		
		B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours	
		AND		
		B.2.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours	
		AND		
		B.2.4 Perform SR 3.2.1.1 ~~and SR 3.2.1.2~~.	72 hours	1
		AND		
		B.2.5 Perform SR 3.2.2.1.	72 hours	
		AND		
		B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days	
DOC M02	C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours	
3.1.3.1 ACTION b	D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR. OR	1 hour	

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.3.1 ACTION b		D.1.2 Initiate boration to restore required SDM to within limit. AND	1 hour
		D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.3.1.1	SR 3.1.4.1	Verify individual rod positions within alignment limit.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.1.3.1.2	SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
3.1.3.4, 4.1.3.4	SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ ~~[2.2]~~ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head	3

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

3.1.3.1 LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

3.1.3.1 Applicability, 3.1.3.4 Applicability

APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.3.1 ACTION a, 4.1.1.1.1, 4.1.1.2, DOC M04	A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Be in MODE 3.	1 hour 1 hour 6 hours
3.1.3.1 ACTION c	B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits. OR B.2.1.1 Verify SDM to be within the limits specified in the COLR. OR	1 hour 1 hour

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.3.1 ACTION c		B.2.1.2 Initiate boration to restore SDM to within limit. AND B.2.2 Reduce THERMAL POWER to ≤ 75% RTP. AND B.2.3 Verify SDM is within the limits specified in the COLR. AND B.2.4 Perform SR 3.2.1.1 ~~and SR 3.2.1.2~~. AND B.2.5 Perform SR 3.2.2.1. AND B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	1 hour 2 hours Once per 12 hours 72 hours (1) 72 hours 5 days
DOC M02	C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
3.1.3.1 ACTION b	D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR. OR	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.3.1 ACTION b	D.1.2 Initiate boration to restore required SDM to within limit. AND D.2 Be in MODE 3.	1 hour 6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.1.3.1.1	SR 3.1.4.1 Verify individual rod positions within alignment limit.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.1.3.1.2	SR 3.1.4.2 Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
3.1.3.4, 4.1.3.4	SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ ~~[2.2]~~ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head	3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS SR 3.1.4.1 and SR 3.1.4.2 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

~~B 3.3 INSTRUMENTATION~~ 4

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ~~e~~ 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System. 1

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. ~~All units have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. 1

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When ~~c~~Control ~~b~~Bank A reaches a predetermined height in the core, ~~c~~Control ~~b~~Bank B begins to move out with 1

BASES

BACKGROUND (continued)

control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions. 9

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, ~~which~~ that are the ~~Bank~~ Demand Position Indication System (commonly called group step counters) and the ~~Digital~~ Rod Position Indication ~~(DRPI)~~ System. 8 1

The ~~Bank~~ Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group ~~all~~ receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The ~~Bank~~ Demand Position Indication System is considered highly precise (± 1 step or ± ~~e~~ 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. 1 8 1

The ~~DRPI~~ Rod Position Indication System provides ~~a highly accurate~~ an indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. ~~To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy.~~ The ~~DRPI~~ Rod Position Indication System is capable of monitoring rod position within at least ± 12 steps ~~with either full accuracy or half accuracy~~. 1 1

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

1. Specified acceptable fuel design limits; ~~or~~
2. Reactor Coolant System (RCS) pressure boundary integrity; ~~and~~ 2

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 1

~~Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking.~~ A different ~~The second~~ type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn. 1

INSERT 2 1

Two types of analysis are performed in regard to static rod misalignment (Ref. 3 ~~4~~). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod and Control ~~from a~~ bank D is fully inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by ± 12 steps. 1

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4 ~~5~~). 1

~~The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.~~ 3

~~Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.~~ 3

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1 **INSERT 1**

There are three RCCA misalignment accidents which are analyzed. They include one or more dropped RCCAs, a dropped RCCA bank, and a statically misaligned RCCA. (Ref. 4)

1 **INSERT 2**

For the dropped RCCA(s) misalignment accident, a negative reactivity insertion will result. For those dropped RCCA(s) that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For the dropped RCCA bank misalignment accident, a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is then tripped. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant.

BASES

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 10% of span ~~24 steps~~ (14.4 ~~15~~ inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. 1

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heat rates (LHRs), or unacceptable SDMs, ~~all of which~~ that may constitute initial conditions inconsistent with the safety analysis. 1 8

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because, except for control rod OPERABILITY testing, the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling. 1

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

BASES

ACTIONS (continued)

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is ~~still~~ trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction. 8

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. ~~For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.~~ 1

Power operation may continue with one RCCA misaligned but trippable (OPERABLE) ~~but misaligned~~, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration. 8 4

BASES

ACTIONS (continued)

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors (F_Q(X,Y,Z) and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y)) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. 1

Reduction of power to 75% RTP ensures that local LHR increases ~~due to~~ resulting from a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. ~~7~~ 5). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. 1 1

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that F_Q(X,Y,Z)~~, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$,~~ and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y) are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate F_Q(X,Y,Z) and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y). 1 1

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in U FSAR Chapter 15 (Ref. 5) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis. 1

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE ~~or Condition~~ in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE ~~2 with K_{eff} < 1.0~~ 3 within 6 hours, which 4 4

BASES

ACTIONS (continued)

obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases ~~or~~ of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. 9

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic ~~bank~~ sequencing of the control banks would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE ~~2 with K_{eff} < 1.0~~ 3 within 6 hours. 1 4

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE ~~2 with K_{eff} < 1.0~~ 3 from full power conditions in an orderly manner and without challenging plant systems. 4

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

~~[~~ Verification that individual rod positions are within alignment limits ~~at a Frequency of 12 hours provides a history that~~ allows the operator to detect a rod that is beginning to deviate from its expected position. ~~The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.~~ 5

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 6 5

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 ~~with $K_{eff} \geq 1.0$~~, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by greater than or equal to 10 steps in either direction will not cause radial or axial power tilts, or oscillations, to occur. ~~[The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods.~~ 4 1 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 6 5

Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head ~~removal~~ installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. INSERT 3 1 1

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, ~~Chapter [15]~~ Section 15.2.3. U 1 7
4. FSAR, ~~Chapter [15]~~ Section 15.4.2. 1 7
5. U FSAR, Chapter ~~[~~15~~]~~. 1 7
6. ~~FSAR, Chapter [15].~~ 1
7. ~~FSAR, Chapter [15].~~ 1

1 **INSERT 3**

Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.

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B 3.1 REACTIVITY CONTROL SYSTEMS
~~B 3.3 INSTRUMENTATION~~ 4

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ~~e~~ 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System. 1

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. ~~All units have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. 1

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When Control Bank A reaches a predetermined height in the core, Control Bank B begins to move out with 1

BASES

BACKGROUND (continued)

control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions. 9

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, ~~which~~ that are the ~~Bank~~ Demand Position Indication System (commonly called group step counters) and the ~~Digital~~ Rod Position Indication ~~(DRPI)~~ System. 8 1

The ~~Bank~~ Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group ~~all~~ receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The ~~Bank~~ Demand Position Indication System is considered highly precise (± 1 step or ± ~~e~~ 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. 1 8 1

The ~~DRPI~~ Rod Position Indication System provides ~~a highly accurate~~ an indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. ~~To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy.~~ The ~~DRPI~~ Rod Position Indication System is capable of monitoring rod position within at least ± 12 steps ~~with either full accuracy or half accuracy~~. 1 1

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

1. Specified acceptable fuel design limits; or
2. Reactor Coolant System (RCS) pressure boundary integrity; and 2

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 1

~~Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking.~~ 1

A different ~~The second~~ type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

INSERT 2 1

Two types of analysis are performed in regard to static rod misalignment (Ref. 3 ~~4~~). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod ~~from a~~ and Control bank D is fully inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by ± 12 steps. 1

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. ~~5~~ 4). 1

~~The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.~~ 3

~~Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.~~ 3

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

INSERT 1

There are three RCCA misalignment accidents which are analyzed. They include one or more dropped RCCAs, a dropped RCCA bank, and a statically misaligned RCCA. (Ref. 4)

1

INSERT 2

For the dropped RCCA(s) misalignment accident, a negative reactivity insertion will result. For those dropped RCCA(s) that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For the dropped RCCA bank misalignment accident, a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is then tripped. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant.

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BASES

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is ~~24 steps~~ 10% of span (~~15~~ 14.4 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. 1

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heat rates (LHRs), or unacceptable SDMs, ~~all of which~~ that may constitute initial conditions inconsistent with the safety analysis. 1 8

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because, except for control rod OPERABILITY testing, the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling. 1

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

BASES

ACTIONS (continued)

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is ~~still~~ trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction. 8

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. ~~For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.~~ 1

Power operation may continue with one RCCA misaligned but trippable (OPERABLE) ~~but misaligned~~, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration. 8 4

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BASES

ACTIONS (continued)

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors (F_Q(X,Y,Z) and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y)) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. 1

Reduction of power to 75% RTP ensures that local LHR increases ~~due to~~ resulting from a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. ~~7~~ 5). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. 1 1

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that F_Q(X,Y,Z)~~, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$,~~ and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y) are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate F_Q(X,Y,Z) and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X,Y). 1 1

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in UFSAR Chapter 15 (Ref. 5) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis. 1

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE ~~or Condition~~ in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE ~~2 with K_{eff} < 1.0~~ 3 within 6 hours, which 4 4

BASES

ACTIONS (continued)

obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of ~~or~~ LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. 9

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic ~~bank~~ sequencing of the control banks would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 ~~2 with K_{eff} < 1.0~~ within 6 hours. 1 4

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 ~~2 with K_{eff} < 1.0~~ from full power conditions in an orderly manner and without challenging plant systems. 4

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

~~[~~ Verification that individual rod positions are within alignment limits ~~at a Frequency of 12 hours provides a history that~~ allows the operator to detect a rod that is beginning to deviate from its expected position. ~~The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.~~ 5

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 6 5

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 ~~with $K_{eff} \geq 1.0$~~, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by greater than or equal to 10 steps in either direction will not cause radial or axial power tilts, or oscillations, to occur. ~~[The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods.~~ 4 1 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 6 5

Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head ~~removal~~ installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. INSERT 3 1 1

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46.

3. FSAR, ~~Chapter [15]~~ Section 15.2.3. U 1 7

4. FSAR, ~~Chapter [15]~~ Section 15.4.2. 1 7

5. FSAR, Chapter [15~~]~~. U 1 7

~~6. FSAR, Chapter [15].~~ 1

~~7. FSAR, Chapter [15].~~ 1

1 **INSERT 3**

Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4 BASES, ROD GROUP ALIGNMENT LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. ISTS B 3.1.4 Applicable Safety Analyses section contains discussion of the Required Action when the LCO is not met. ITS B 3.1.4 Applicable Safety Analyses section does not contain this discussion. This information is adequately addressed in the Bases for ACTIONS

4. Changes are made to be consistent with the Specification.

5. ISTS SR 3.1.4.1 and SR 3.1.4.2 Bases provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

7. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

8. Editorial changes made for enhanced clarity/consistency.

9. Typographical/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

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

A01

ITS — ITS 3.1.5

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ~~ROD~~ BANK INSERTION LIMIT — A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 — 3.1.3.5 ~~All~~ Each shutdown ~~rods~~ bank shall be limited in physical insertion as specified in the COLR.

Applicability — APPLICABILITY: MODES 1~~*~~ and 2~~*#~~ — A03, M01

ACTION:

ACTION B — Applicability Note
a. With a ~~maximum of one shutdown rod~~ one or more shutdown banks inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2 ~~or when complying with ACTION b of this specification~~, within ~~one~~ two hour either: — L01

Add proposed Required Actions B.1.1 and B.1.2

1. Restore the ~~rod~~ bank to within the insertion limit specified in the COLR, or
2. ~~Declare the rod to be inoperable and apply ACTION 3.1.3.1.c.3.~~

Add proposed ACTION C

ACTION A
b. With a maximum of one shutdown bank inserted beyond the insertion limit specified in the COLR ~~during surveillance testing pursuant to Specification 4.1.3.1.2~~ and immovable due to malfunctions in the rod control system, POWER OPERATION may continue provided that: — A04

1. The shutdown bank is inserted no more than 18 steps below the insertion limit as measured by the group step counter demand position indicators,
2. ~~The affected bank is trippable~~, — Each control and shutdown rod within the limits of LCO 3.1.4 — A05
3. ~~Each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position,~~
4. The insertion limits of Specification 3.1.3.6 are met for each control bank,
5. No reactor coolant system boron concentration dilution activities or power level increases are allowed,
6. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined to be met at least once per 12 hours or upon insertion of the controlling bank more than 5 steps from the initial position, and
7. The shutdown bank is restored to within the insertion limit specified in the COLR within 72 hours.

ACTION C — Otherwise, be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1
4.1.3.5 Each shutdown ~~rod~~ bank shall be determined to be within the insertion limit specified in the COLR: — A02

a. ~~Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and~~ — L02

b. ~~At least once per 12 hours thereafter~~ In accordance with the Surveillance Frequency Control Program. — LA01

~~*See Special Test Exceptions 3.10.2 and 3.10.3.~~ — A03

~~#With K_{eff} greater than or equal to 1.0.~~ — M01

ITS A01 ITS 3.1.5

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ~~ROD~~ BANK INSERTION LIMIT A02

LIMITING CONDITION FOR OPERATION A02

LCO 3.1.5 3.1.3.5 ~~All~~ Each shutdown ~~rods~~ bank shall be limited in physical insertion as specified in the COLR: A03

Applicability APPLICABILITY: Modes 1~~*~~ and 2~~*#~~. M01

ACTION:

ACTION B Applicability Note

a. With a ~~maximum of one shutdown rod~~ one or more shutdown banks inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2 ~~or when complying with ACTION b of this specification~~, within ~~one~~ two hour either: L01

Add proposed Required Actions B.1.1 and B.1.2

1. Restore the ~~rod~~ bank to within the insertion limit specified in the COLR, or

~~2. Declare the rod to be inoperable and apply ACTION 3.1.3.1.c.3.~~

Add proposed ACTION C

ACTION A

b. With a maximum of one shutdown bank inserted beyond the insertion limit specified in the COLR ~~during surveillance testing pursuant to Specification 4.1.3.1.2~~ and immovable due to malfunctions in the rod control system, POWER OPERATION may continue provided that: A04

1. The shutdown bank is inserted no more than 18 steps below the insertion limit as measured by the group step counter demand position indicators,
2. ~~The affected bank is trippable,~~ Each control and shutdown rod within the limits of LCO 3.1.4 A05
3. ~~Each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position,~~
4. The insertion limits of Specification 3.1.3.6 are met for each control bank,
5. No reactor coolant system boron concentration dilution activities or power level increases are allowed,
6. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined to be met at least once per 12 hours or upon insertion of the controlling bank more than 5 steps from the initial position, and
7. The shutdown bank is restored to within the insertion limit specified in the COLR within 72 hours.

ACTION C Otherwise, be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown ~~rod~~ bank shall be determined to be within the insertion limit specified in the COLR: A02

SR 3.1.5.1

~~a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and~~ L02

b. ~~At least once per 12 hours thereafter.~~ In accordance with the Surveillance Frequency Control Program LA01

~~* See Special Test Exceptions 3.10.2 and 3.10.3.~~ A03

~~# With K_{eff} greater than or equal to 1.0~~ M01

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.1.3.5 states "All shutdown rods shall be limited in physical insertion as specified in the COLR. Additionally, the title of CTS 3.1.3.5 is "SHUTDOWN ROD INSERTION LIMIT." ITS LCO 3.1.5 states "Each shutdown bank shall be within insertion limits specified in the COLR." Furthermore, ITS 3.1.5 title has been changed to "SHUTDOWN BANK INSERTION LIMIT." This changes the CTS by referring to each bank instead of all rods.

The purpose of CTS 3.1.3.5 is to ensure that sufficient negative reactivity is available to shutdown the reactor and to maintain the SDM. This change is acceptable because the requirements have not changed. ITS 3.1.5 will continue to ensure that sufficient negative reactivity is available to shutdown the reactor and to maintain the SDM. This change is a change in presentation to match the ISTS format. Therefore, this change is designated as an administrative change because it does not result in a technical change to the CTS.

A03 CTS 3.1.3.5 Applicability is modified by a footnote (footnote *) which states "See Special Test Exceptions 3.10.2 and 3.10.3." ITS 3.1.5 Applicability does not contain this footnote or a reference to the Special Test Exceptions. This changes the CTS by not including footnote *.

The purpose of Footnote * is to alert the Technical Specification user that a Special Test Exception exists that may modify the Applicability of this Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.1.3.5 ACTION b states that POWER OPERATION may continue with a maximum of one shutdown bank inserted beyond the insertion limit specified in the COLR during surveillance testing pursuant to Specification 4.1.3.1.2 and immovable due to malfunctions in the rod control system. ITS 3.1.5 ACTION A allows POWER OPERATION to continue with one shutdown bank inserted beyond the insertion limit and immovable due to malfunctions in the rod control system. This changes the CTS by removing the qualification statement "during surveillance testing pursuant to Specification 4.1.3.1.2."

The purpose of CTS 3.1.3.5 ACTION b is to allow time for diagnosis and repair of an inoperable shutdown bank if the failure is external to the control rod drive mechanism. Since the shutdown banks are required to be fully withdrawn in

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

MODES 1 and 2, the only time the shutdown banks are inserted, in these MODES, are during the performance of the rod freedom of movement test of CTS 4.1.3.1.2 and low power physics testing. Therefore, the statement "during surveillance testing pursuant to Specification 4.1.3.1.2" is not necessary. Furthermore, ITS LCO 3.1.5 is not applicable during the rod freedom of movement test, as stated in the ITS 3.1.5 Applicability Note. Therefore, referencing the SR (ITS SR 3.1.4.2) within the Specification would be confusing. This change is designated as administrative because it does not result in a technical change to the specifications.

A05 CTS 3.1.3.5 ACTION b states, in part, that with a maximum of one shutdown bank inserted beyond the insertion limit, POWER OPERATION may continue provided that the affected bank is trippable and each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position. ITS 3.1.5 Required Action A.2 requires immediate verification that each control and shutdown rod are within the limits of LCO 3.1.4. This changes the CTS by specifically stating that the control and shutdown banks shall be within the limits of LCO 3.1.4.

The purpose of this portion of CTS 3.1.3.5 ACTION b is to verify the requirements of CTS 3.1.3.1 are met. CTS 3.1.3.1 states that all full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position. In CTS 3.1.3.5 ACTION b, verifying that the affected bank is trippable, is verifying that the bank is OPERABLE. Additionally, verifying each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position in CTS 3.1.3.5, is the same as verifying the shutdown and control rods are positioned within ± 12 steps (indicated position) of their group step counter demand position. The ITS 3.1.5 Required Action B.2 statement eliminates any confusion as to what actions are being taken. This change is designated as administrative because it does not result in a technical change to the specifications.

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.5 is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$. MODE 2 is modified by CTS 3.1.3.5 footnote #. ITS 3.1.5 is applicable in MODES 1 and 2. This changes the CTS by expanding the Applicability from MODE 2 with the reactor critical to all of MODE 2.

The purpose of CTS 3.1.3.5 is to ensure that the shutdown banks are fully withdrawn prior to withdrawing the control banks in order to ensure that there is sufficient shutdown margin available to quickly shutdown the reactor. This change is acceptable because applying the requirement prior to removing the control banks and bringing the reactor critical ensures that the shutdown margin is available and is consistent with plant operation, in that the shut down banks are completely withdrawn before beginning to withdraw the control banks and approaching criticality. This change is designated as more restrictive because it

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DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

increases the conditions under which Technical Specification controls will be applied.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.1.3.5.b requires verification that each shutdown rod is within the insertion limit specified in the COLR at least once per 12 hours. ITS 3.1.5.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.5 ACTION a provides compensatory actions for a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR. The actions require within one hour either restore the rod to within the insertion limit specified in the COLR or declare the rod to be inoperable and apply ACTION 3.1.3.1.c.3. For more than one shutdown rod beyond the insertion limit, CTS 3.1.3.5 does not contain a specific requirement; therefore, entry into CTS 3.0.3 is required. ITS 3.1.5 ACTION B provides Required Actions for one or more shutdown banks not within limits. ITS 3.1.5 Required Action B.1 requires either verification the SDM is within the limits specified in the COLR (Required Action B.1.1) or the initiation of boration to restore SDM to within limits (Required Action B.1.2), both within 1 hour. ITS 3.1.5 Required Action B.2 requires restoration of the shutdown banks to

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

within limits within 2 hours. Additionally, ITS 3.1.5 ACTION C requires if any Required Action and associated Completion Time is not met, the unit must be in MODE 3 within 6 hours. This changes the CTS by allowing more than one shutdown rod to be outside the insertion limits specified in the COLR, provides an additional hour to restore the shutdown bank or shutdown rod to within limits, eliminates the allowance to declare the rod inoperable and to take the ACTIONS of Specification 3.1.3.1, and adds the requirement to verify SDM or to initiate boration within one hour. It also eliminates the requirement to enter CTS 3.0.3 if more than one shutdown rod is inserted beyond the insertion limits.

The purpose of CTS 3.1.3.5 ACTION a is to ensure the shutdown banks are fully withdrawn in order to ensure that there is sufficient margin available to quickly shutdown the reactor. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering that only a small amount of time is provided to establish the required features, and the low probability of a DBA occurring during the repair period. Allowing an additional hour to restore one or more shutdown banks (or more than one shutdown rod) inserted below the insertion limit is appropriate as it may avoid a shutdown, a unit transient, while the rod control system is not in full working order. The ITS requires verification that the shutdown margin requirement is met or actions to restore the shutdown margin to within its limit within 1 hour, so all safety analysis assumptions are being met. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.5.a requires verification that each shutdown rod is within the insertion limit specified in the COLR within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality. ITS 3.1.5 does not require verification that the shutdown rods are above the insertion limits within 15 minutes prior to control bank withdrawal. This changes the CTS by eliminating the requirement that the shutdown banks be verified to be above the insertion limit within 15 minutes prior to withdrawing control banks A, B, C, and D.

The purpose of CTS 4.1.3.5.a is to verify the shutdown rods are withdrawn above the insertion limit prior to withdrawing the control banks. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the equipment being used to meet the LCO can perform its required function. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence the equipment can perform its assumed safety function. Under the ITS Applicability of MODE 2 and the requirement of ITS LCO 3.0.4, the shutdown banks must be above the insertion limit prior to entering the ITS Applicability of MODE 2. However, it is not required to verify compliance within a specified time prior to initial control bank withdrawal. Specifying a time is not necessary to ensure the shutdown banks are above the insertion limit prior to initial control bank withdrawal as long as the shutdown banks are withdrawn before withdrawing the control banks. This change is

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

designated as less restrictive because a Surveillance which was required in CTS will not be required in the ITS.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

3.1.3.5 LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

Applicability APPLICABILITY: MODES 1 and 2.

-----NOTE-----

ACTION a This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 1		
~~A~~ B. One or more shutdown banks not within limits for reasons other than Condition A.	~~A~~ B.1.1 Verify SDM is within the limits specified in the COLR. OR ~~A~~ B.1.2 Initiate boration to restore SDM to within limit. AND ~~A~~ B.2 Restore shutdown bank~~s~~(s) to within limits.	1 hour 1 1 hour 1 2 hours 2 1
~~B~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 Be in MODE 3.	6 hours 1

ACTION a (beside Condition A); ACTION b, DOC L01 (beside Condition B)

1 **INSERT 1**

ACTION b	A. One shutdown bank not within limits and immovable due to malfunctions in the Rod Control System.	A.1 Verify shutdown bank is inserted ≤ 18 steps below the insertion limit as measured by group step counter demand position indicators. AND A.2 Verify each control and shutdown rod is within limits of LCO 3.1.4, "Rod Group Alignment Limits." AND A.3 Verify each control bank is within insertion limits of LCO 3.1.6, "Rod Group Insertion Limits.". AND A.4 Verify no Reactor Coolant System boron dilution activities in progress. AND A.5 Verify no power level increases. AND A.6 Verify SDM is within limits specified in the COLR. AND A.7 Restore shutdown bank to within limits.	Immediately Immediately Immediately Immediately Immediately Once per 12 hours AND Immediately upon insertion of controlling bank more than 5 steps from the initial position 72 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.1.3.5	SR 3.1.5.1 Verify each shutdown bank is within the insertion limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

3.1.3.5 LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

Applicability APPLICABILITY: MODES 1 and 2.

--NOTE--

ACTION a This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ~~A~~ B. One or more shutdown banks not within limits for reasons other than Condition A.	~~A~~ B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour (1)
	OR	
	~~A~~ B.1.2 Initiate boration to restore SDM to within limit.	1 hour (1)
	AND	
	~~A~~ B.2 Restore shutdown bank~~s~~ (s) to within limits.	2 hours (2) (1)
ACTION b, DOC L01 ~~B~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 Be in MODE 3.	6 hours (1)

1 **INSERT 1**

ACTION b	A. One shutdown bank not within limits and immovable due to malfunctions in the Rod Control System.	A.1	Verify shutdown bank is inserted ≤ 18 steps below the insertion limit as measured by group step counter demand position indicators.	Immediately
		AND		
		A.2	Verify each control and shutdown rod is within limits of LCO 3.1.4, "Rod Group Alignment Limits."	Immediately
		AND		
		A.3	Verify each control bank is within insertion limits of LCO 3.1.6, "Rod Group Insertion Limits.".	Immediately
		AND		
		A.4	Verify no Reactor Coolant System boron dilution activities in progress.	Immediately
		AND		
		A.5	Verify no power level increases.	Immediately
		AND		
		A.6	Verify SDM is within limits specified in the COLR.	Once per 12 hours AND Immediately upon insertion of controlling bank more than 5 steps from the initial position
		AND		
		A.7	Restore shutdown bank to within limits.	72 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.1.3.5 SR 3.1.5.1	Verify each shutdown bank is within the insertion limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

1. ISTS 3.1.5 has been modified to include a new ACTION (ITS 3.1.5 ACTION A). ITS 3.1.5 requires entering Condition A when one shutdown bank is inserted beyond the insertion limit and immovable due to a malfunction in the rod control system. ITS 3.1.5 Required Action A.1 requires an immediate verification that the shutdown bank is inserted less than or equal to 18 steps below the insertion limit as measured by the group step counter demand position indicators. ITS 3.1.5 Required Action A.2 requires an immediate verification that each control and shutdown rod is within the limits of LCO 3.1.4. ITS 3.1.5 Required Action A.3 requires an immediate verification that each control bank is within the insertion limits of LCO 3.1.6. ITS 3.1.5 Required Action A.4 requires an immediate verification that there are no reactor coolant system boron dilution concentration activities. ITS 3.1.5 Required Action A.5 requires an immediate verification that there are no power level increases. ITS 3.1.5 Required Action A.6 requires verification that the SDM is within the limits specified in the COLR once per 12 hours and upon insertion of the controlling bank more than 5 steps from the initial position. ITS 3.1.5 Required Action A.7 requires the restoration of the shutdown bank to within limits in 72 hours. This addition is acceptable because it reflects the current licensing basis. Furthermore, ISTS 3.1.5 Condition A (ITS 3.1.5 Condition B) was modified to state it is applicable for reasons other than Condition A, consistent with current licensing. This change was approved in License Amendment 215 for Unit 1 and License Amendment 205 for Unit 2 (ADAMS Accession No. ML013330266). Additionally, due to the addition of ITS 3.1.5 ACTION A, the subsequent ACTIONS were renumbered.

2. Editorial changes made for enhanced clarity/consistency.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. ISTS SR 3.1.5.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements. 1

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The

BASES

BACKGROUND (continued)

shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. INSERT 1 The shutdown banks must be ~~completely~~ withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. ~~The shutdown banks are then left in this position until the reactor is shut down.~~ They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. 1 1

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

   1. Specified acceptable fuel design limits or

   2. RCS pressure boundary integrity and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analyses (e) involving core reactivity and SDM (Ref. 3). 2

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1 **INSERT 1**

They are moved quarterly or following maintenance to ensure trippability but are returned to the withdrawn position when the testing is completed.

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BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks , except for control rod OPERABILITY testing, are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODE 2 k_{eff} < 1.0, MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6. 1 3

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

INSERT 2

~~A~~B.1.1, ~~A~~B.1.2, and ~~A~~B.2 3

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

~~B~~C.1 3

If INSERT 3 ~~the shutdown banks cannot be restored to within their insertion limits within 2 hours~~, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. 3

2 **INSERT 2**

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one shutdown bank is inserted beyond the insertion limit and is immovable due to a malfunction in the rod control system, 72 hours are provided to restore the shutdown banks to within limits. Additionally, immediate verification is required to prove that the shutdown bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual control rod alignment limits of LCOs 3.1.4 and 3.1.6 are met, there are no reactor coolant system boron dilution activities, and there are no power level increases are taking place. Furthermore, a verification of SDM is required within 12 hours or when the controlling banks are inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.6 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted rod. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Conditions are not met, Condition C must be applied.

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system.

2 **INSERT 3**

the Required Action(s) of Condition A or B are not met within the associated Completion Times

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

~~[Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~-----]~~ 4

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

3. U FSAR, Chapter ~~[~~15~~]~~. 1 6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements. 1

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The

BASES

BACKGROUND (continued)

shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. INSERT 1 The shutdown banks must be ~~completely~~ withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. ~~The shutdown banks are then left in this position until the reactor is shut down.~~ They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. 1 1

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

   1. Specified acceptable fuel design limits or

   2. RCS pressure boundary integrity and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analys~~i~~es involving core reactivity and SDM (Ref. 3). 2

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1 **INSERT 1**

They are moved quarterly or following maintenance to ensure trippability but are returned to the withdrawn position when the testing is completed.

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BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks , except for control rod OPERABILITY testing, are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODE 2 k_{eff} < 1.0, MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6. 1 3

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

INSERT 2

~~A~~ B.1.1, ~~A~~ B.1.2, and ~~A~~ B.2 3

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

3

~~B~~ C.1

If INSERT 3 ~~the shutdown banks cannot be restored to within their insertion limits within 2 hours~~, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. 3

---

2 **INSERT 2**

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one shutdown bank is inserted beyond the insertion limit and is immovable due to a malfunction in the rod control system, 72 hours are provided to restore the shutdown banks to within limits. Additionally, immediate verification is required to prove that the shutdown bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual control rod alignment limits of LCOs 3.1.4 and 3.1.6 are met, there are no reactor coolant system boron dilution activities, and there are no power level increases are taking place. Furthermore, a verification of SDM is required within 12 hours or when the controlling banks are inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.6 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted rod. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Conditions are not met, Condition C must be applied.

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system.

2 **INSERT 3**

the Required Action(s) of Condition A or B are not met within the associated Completion Times

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

~~[Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

4 5 4

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. U FSAR, Chapter ~~[~~15~~]~~. 1 6

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5 BASES, SHUTDOWN BANK INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Editorial changes made for enhanced clarity/consistency.

3. Changes are made to be consistent with changes made to the Specification. Additionally, the subsequent ACTIONS have been renumbered.

4. ISTS SR 3.1.5.1 and SR 3.1.5.2 Bases provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ITS 3.1.6, CONTROL BANK INSERTION LIMITS

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

A01

ITS ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

BANK A01

CONTROL ~~ROD~~ INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

, sequence, and overlap limits M01

LCO 3.1.6 3.1.3.6 The control banks shall be limited in physical insertion as specified in the COLR.

A02

Applicability APPLICABILITY: MODES 1~~*~~ and 2~~*~~#.

ACTION:

ACTION B

a. With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2 or when complying with ACTION b of this specification, either:

Applicability Note

Add proposed Required Action B.1.1 and B.1.2 M02

1. Restore the control banks to within the limits within two hours, or

~~2. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR, or~~ A03

Add proposed ACTION C M01

ACTION D 3. Be in ~~HOT STANDBY~~ within 6 hours.

MODE 2 with k_{eff} < 1.0 A04

b. With a maximum of one control bank inserted beyond the insertion limit specified in the COLR ~~during surveillance testing pursuant to Specification 4.1.3.1.2~~ and immovable due to malfunctions in the rod control system, POWER OPERATION## may continue provided that: A05

1. The control bank is inserted no more than 18 steps below the insertion limit as measured by the group step counter demand position indicators,

Each control and shutdown rod within the limits of LCO 3.1.4. A06

2. ~~The affected bank is trippable,~~

3. ~~Each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position,~~

ACTION A

4. The insertion limits of Specification 3.1.3.5 are met for each shutdown bank,
5. No reactor coolant system boron concentration dilution activities or power level increases are allowed,
6. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined to be met at least once per 12 hours or upon insertion of the controlling bank more than 5 steps from the initial position, and
7. The control bank is restored to within the insertion limit specified in the COLR within 72 hours.

ACTION D Otherwise, be in ~~HOT STANDBY~~ within the next 6 hours.

MODE 2 with k_{eff} < 1.0 A04

SURVEILLANCE REQUIREMENTS

In accordance with the Surveillance Frequency Control Program LA01

SR 3.1.6.2 4.1.3.6 The position of each control bank shall be determined to be within the insertion limits ~~at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours~~.

L01

Add proposed SR 3.1.6.3 M01

~~*See Special Test Exceptions 3.10.2 and 3.10.3.~~

A02

Applicability #With K_{eff} greater than or equal to 1.0.

ACTION A Note ## Provision for continued POWER OPERATION does not apply to the controlling bank(s) ~~(normally Control Bank D)~~ inserted beyond the insertion limit.

LA02

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A01

ITS ITS 3.1.6

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} Greater Than 200°F

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

Applicability APPLICABILITY: MODES 1, 2*, 3, and 4. See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

See ITS Chapter 1.0

In accordance with the Surveillance Frequency Control Program LA01

SR 3.1.6.2 b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, ~~at least once per 12 hours~~ by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

SR 3.1.6.1 c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

See ITS 3.1.1

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A01

ITS ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

CONTROL ~~ROD~~ BANK INSERTION LIMITS A01

LIMITING CONDITION FOR OPERATION

LCO 3.1.6 3.1.3.6 The control banks shall be limited in physical insertion , sequence, and overlap limits as specified in the COLR M01

Applicability APPLICABILITY: Modes 1~~*~~ and 2~~*~~#. A02

ACTION:

ACTION B a. With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2 or (Applicability Note) when complying with ACTION b of this specification, either:

Add proposed Required Action B.1.1 and B.1.2 M02

1. Restore the control banks to within the limits within two hours, or

~~2. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR, or~~ A03

Add proposed ACTION C M01

3. Be in ~~HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 within 6 hours. A04

ACTION D

ACTION A b. With a maximum of one control bank inserted beyond the insertion limit specified in the COLR ~~during surveillance testing pursuant to Specification 4.1.3.1.2 and~~ immovable due to malfunctions in the rod control system, POWER OPERATION## may continue provided that: A05

1. The control bank is inserted no more than 18 steps below the insertion limit as measured by the group step counter demand position indicators,
2. ~~The affected bank is trippable,~~
3. ~~Each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position,~~ Each control and shutdown rod within the limits of LCO 3.1.4 A06
4. The insertion limits of Specification 3.1.3.5 are met for each shutdown bank,
5. No reactor coolant system boron concentration dilution activities or power level increases are allowed,
6. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined to be met at least once per 12 hours or upon insertion of the controlling bank more than 5 steps from the initial position, and
7. The control bank is restored to within the insertion limit specified in the COLR within 72 hours.

Otherwise, be in ~~HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 within the next 6 hours. A04

SURVEILLANCE REQUIREMENTS

SR 3.1.6.2 4.1.3.6 The position of each control bank shall be determined to be within the insertion limits ~~at least once per~~ In accordance with the Surveillance Frequency Control Program ~~12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours~~. LA01 L01

Add proposed SR 3.1.6.3 M01

~~* See Special Test Exceptions 3.10.2 and 3.10.3.~~ A02

Applicability # With K_{eff} greater than or equal to 1.0.

ACTION A Note ## Provision for continued POWER OPERATION does not apply to the controlling bank(s) ~~(normally Control Bank D)~~ inserted beyond the insertion limit. LA02

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A01

ITS

ITS 3.1.6

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \geq 200°F$

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for 4 loop operation.

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

See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

See ITS Chapter 1.0

In accordance with the Surveillance Frequency Control Program

LA01

SR 3.1.6.2

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, ~~at least once per 12 hours~~ by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

SR 3.1.6.1

c. When in MODE 2, with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

See ITS 3.1.1

* See Special Test Exception 3.10.1

See ITS 3.1.1

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DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.1.3.6 Applicability is modified by a footnote (footnote *) that states "See Special Test Exceptions 3.10.2 and 3.10.3." ITS 3.1.6 Applicability does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS by not including footnote *.

The purpose of Footnote * is to alert the Technical Specification user that a Special Test Exception exists that may modify the Applicability of this Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.1.3.6 ACTION a states that with the control banks beyond the insertion limits, to restore the control bank to within limits within 2 hours or reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR. ITS 3.1.6 Required Action B.2 requires restoring the control banks to within limits within 2 hours. This changes the CTS by eliminating the explicit statement that compliance with the LCO can be restored in order to exit the ACTION.

This change is acceptable because the requirements have not changed. When THERMAL POWER is reduced, the insertion limits, which are a function of power, are lowered. When the insertion limits are lowered, the control banks, which were previously inserted below the insertion limits, will then come within the new limit. This is the same as the CTS ACTION a option to restore the control banks to within the limit. This change is considered administrative because the technical requirements have not changed.

A04 CTS 3.1.3.6 ACTION a.3 and ACTION b require the unit to be in HOT STANDBY (MODE 3) within 6 hours if ACTION a or b are not met. The CTS Applicability is MODES 1 and 2 with $k_{eff} \geq 1.0$. ITS 3.1.6 ACTION D requires the unit to be in MODE 2 with $k_{eff} < 1.0$. This changes the CTS by requiring the unit to be in MODE 2 with $k_{eff} < 1.0$ instead of HOT STANDBY (MODE 3).

This change is acceptable because the requirements have not changed. In the CTS, ACTIONS are only required to be followed while in the Mode of Applicability. The CTS control bank insertion limits are applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$. Therefore, under the CTS, the unit does not have to enter

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DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

MODE 3 because the Applicability of the LCO has been exited when in MODE 2 with k_{eff} < 1.0. As a result, there is no difference between the CTS and the ITS requirements. This change is designated as administrative because it does not result in a technical change to the CTS.

A05 CTS 3.1.3.6 ACTION b states that POWER OPERATION may continue with a maximum of one control bank inserted beyond the insertion limit specified in the COLR during surveillance testing pursuant to Specification 4.1.3.1.2 and immovable resulting from malfunctions in the rod control system. ITS 3.1.6 ACTION A allows, in part, POWER OPERATION to continue with one control bank inserted beyond the insertion limit and immovable. This changes the CTS by removing the qualification statement "during surveillance testing pursuant to Specification 4.1.3.1.2."

The purpose of CTS 3.1.3.6 ACTION b is to allow time for diagnosis and repair to an inoperable control bank if the failure is external to the control rod drive mechanism. Since the shutdown banks are required to be fully withdrawn in MODES 1 and 2, the only time the control banks are inserted, in these MODES, are during the performance of the rod freedom test of CTS 4.1.3.1.2. Therefore, the statement "during surveillance testing pursuant to Specification 4.1.3.1.2" is not necessary. Furthermore, ITS LCO 3.1.6 is not applicable during the rod freedom test, as stated in the ITS 3.1.6 Applicability Note. Therefore, referencing the SR (ITS SR 3.1.4.2) within the Specification would be confusing. This change is designated as administrative because it does not result in a technical change to the specifications.

A06 CTS 3.1.3.6 ACTION b states, in part, that with a maximum of one control bank inserted beyond the insertion limit, POWER OPERATION may continue provided that the affected bank is trippable and each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position. ITS 3.1.6 Required Action A.2 requires, in part, verification that each control and shutdown rod is within the limits of LCO 3.1.4. This changes the CTS by specifically stating that the control and shutdown rods shall be verified to be within the limits of LCO 3.1.4.

The purpose of this portion of CTS 3.1.3.6 ACTION b is to verify the requirements of CTS 3.1.3.1 are met. CTS 3.1.3.1 states that all full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position. In CTS 3.1.3.6 ACTION b, verifying that the affected bank is trippable, is verifying that the bank is OPERABLE. Additionally, when the control rod is aligned to within ± 12 steps of its respective group step counter demand position in CTS 3.1.3.6, this is the same as verifying the shutdown and control rods are positioned within ± 12 steps (indicated position) of their group step counter demand position. The ITS 3.1.6 Required Action A.2 statement eliminates any confusion as to what actions are being taken. This change is designated as administrative because it does not result in a technical change to the specifications.

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DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.6 requires the control banks to be limited in physical insertion as specified in the COLR. ITS LCO 3.1.6 requires the control banks to be within insertion, sequence and overlap limits specified in the COLR. ITS 3.1.6 ACTION C provides requirements when not meeting the sequence and overlap requirements. ITS SR 3.1.6.3 requires verification of the sequence and overlap limits every 12 hours. This changes the CTS by adding the requirements on the sequence and overlap limits in addition to the Technical Specifications.

This change is acceptable because the control bank sequence and overlap limits are important assumptions in the core power distribution analyses. The addition of these requirements, ACTIONS, and Surveillance Requirements provides assurance that the core power distribution is maintained within the design predictions. This change is designated as more restrictive because new requirements are added to the CTS.

M02 CTS 3.1.3.6 ACTION a requires, in part, control banks inserted beyond the insertion limits to be restored within 2 hours. ITS 3.1.6 ACTION B contains the same requirements and adds the requirement to either verify the SDM is within limits or initiate boration to restore SDM to within limits within one hour. This changes the CTS by adding the requirement to verify SDM or to initiate boration to restore the SDM within one hour when control banks are below the insertion limits.

This change is acceptable because it verifies that the initial conditions of the accident analyses are maintained. In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, SDM is ensured by adhering to the control and shutdown bank insertion limits. If the control banks are not within their insertion limits, then SDM must be verified to be within limits or actions must be initiated to restore SDM to within limits. This change is designated as more restrictive because requirements are added to the CTS.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.1.3.6 requires, in part, the position of each control bank shall be determined to be within the insertion limits at least once per 12 hours. CTS 4.1.1.1.1.b requires, in part, verifying the control bank withdrawal is within limits of Specification 3.1.3.6 at least once per 12 hours. ITS SR 3.1.6.2 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.6 requires the control banks to be limited in physical insertion as specified in the COLR. CTS 3.1.3.6 ACTION b allows POWER OPERATION to continue with a maximum of one control bank inserted beyond the limit specified in the COLR during the rod freedom of movement surveillance provided the control bank is immovable due to a malfunction of the rod control system and the specified actions are met within the specified times specified. Additionally, footnote ## states the provision for continued POWER OPERATION does not apply to the controlling bank(s) (normally Control Bank D) inserted beyond the insertion limit. ITS LCO 3.1.6 and ACTION A retain the same requirements, but do not specify that Control Bank D is normally the controlling bank. This changes the CTS by relocating the details that Control Bank D is normally the controlling bank to the Bases.

The removal of these details, that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the control banks to be within the insertion limits specified in the COLR, as well as the Actions to take when a control bank is not within the limits specified in the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by

DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.6 requires verification that each control rod is within the insertion limit at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then it requires verification of the individual rod positions at least once per 4 hours. ITS 3.1.6.2 requires verification that each control bank insertion is within the insertion limits specified in the COLR in accordance with the Surveillance Frequency Control Program. This changes the CTS by eliminating the requirement to verify the control bank insertion to be within limits every 4 hours when the Rod Insertion Limit Monitor is inoperable.

The purpose of CTS 4.1.3.6 is to periodically verify that the rods are within the alignment limit specified in the LCO. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of rod position verification when the Rod Insertion Limit Monitor is inoperable is unnecessary because inoperability of the alarm does not increase the possibility that the control banks are inserted below the limits. The Rod Insertion Limit Monitor alarm is for indication only; its use is not credited in any of the safety analyses. This change is designated as less restrictive because a Surveillance which was required in CTS will not be required in the ITS.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

3.1.3.6

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

Applicability, Footnote #

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTION a

---NOTE---
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ~~A~~ B. Control bank insertion limits not met for reasons other than Condition A.	~~A~~ B.1.1 Verify SDM is within the limits specified in the COLR. OR ~~A~~ B.1.2 Initiate boration to restore SDM to within limit. AND ~~A~~ B.2 Restore control bank(s) to within limits.	1 hour 1 hour 2 hours
DOC M01 ~~B~~ C. Control bank sequence or overlap limits not met.	~~B~~ C.1.1 Verify SDM is within the limits specified in the COLR. OR ~~B~~ C.1.2 Initiate boration to restore SDM to within limit. AND	1 hour 1 hour

1 1 1 1 1 1

1

INSERT 1

ACTION b	A. ------------NOTE------------ Only applicable to control bank(s) that are not a controlling bank. --------------------------------- One control bank not within limits and immovable due to malfunctions in the Rod Control System.	A.1 Verify control bank is inserted ≤ 18 steps below the insertion limit as measured by group step demand position indicators.	Immediately
		AND	
		A.2 Verify each control and shutdown rod is within limits of LCO 3.1.4, "Rod Group Alignment Limits."	Immediately
		AND	
		A.3 Verify each shutdown bank is within insertion limits of LCO 3.1.5, "Shutdown Bank Insertion Limits."	Immediately
		AND	
		A.4 Verify no Reactor Coolant System boron dilution activities.	Immediately
		AND	
		A.5 Verify no power level increases.	Immediately
		AND	
		A.6 Verify SDM is within limits specified in the COLR.	Once per 12 hours AND Immediately upon insertion of controlling bank more than 5 steps from the initial position
		AND	
		A.7 Restore control bank to within limits.	72 hours

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
DOC M01		~~B~~ C.2	Restore control bank sequence and overlap to within limits.	2 hours	1
ACTION a.3, ACTION b	~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1	Be in MODE 2 with k_{eff} < 1.0.	6 hours	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.1.1.1.c	SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality	
4.1.3.6, 4.1.1.1.1.b	SR 3.1.6.2	Verify each control bank insertion is within the insertion limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3
DOC M01	SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

3.1.3.6

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

Applicability, Footnote #

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥1.0.

--NOTE--

ACTION a

This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

INSERT 1 (1)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ~~A~~ B. Control bank insertion limits not met for reasons other than Condition A.	~~A~~ B.1.1 Verify SDM is within the limits specified in the COLR. OR ~~A~~ B.1.2 Initiate boration to restore SDM to within limit. AND ~~A~~ B.2 Restore control bank(s) to within limits.	1 hour (1) 1 hour (1) 2 hours (1)
DOC M01 ~~B~~ C. Control bank sequence or overlap limits not met.	~~B~~ C.1.1 Verify SDM is within the limits specified in the COLR. OR ~~B~~ C.1.2 Initiate boration to restore SDM to within limit. AND	1 hour (1) 1 hour (1)

1 **INSERT 1**

ACTION b	A. ------------NOTE------------ Only applicable to control bank(s) that are not a controlling bank. --------------------------------- One control bank not within limits and immovable due to malfunctions in the Rod Control System.	A.1 Verify control bank is inserted ≤ 18 steps below the insertion limit as measured by group step demand position indicators.	Immediately
		AND	
		A.2 Verify each control and shutdown rod is within limits of LCO 3.1.4, "Rod Group Alignment Limits."	Immediately
		AND	
		A.3 Verify each shutdown bank is within insertion limits of LCO 3.1.5, "Shutdown Bank Insertion Limits."	Immediately
		AND	
		A.4 Verify no Reactor Coolant System boron dilution activities.	Immediately
		AND	
		A.5 Verify no power level increases.	Immediately
		AND	
		A.6 Verify SDM is within limits specified in the COLR.	Once per 12 hours AND Immediately upon insertion of controlling bank more than 5 steps from the initial position
		AND	
		A.7 Restore control bank to within limits.	72 hours

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
DOC M01		~~B~~ C.2 Restore control bank sequence and overlap to within limits.	2 hours	1
ACTION a.3, ACTION b	~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 2 with k_{eff} < 1.0.	6 hours	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.1.1.1.c	SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality	
4.1.3.6, 4.1.1.1.1.b	SR 3.1.6.2	Verify each control bank insertion is within the insertion limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3
DOC M01	SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

1. ISTS 3.1.6 has been modified to include a new ACTION (ITS 3.1.6 ACTION A). ITS 3.1.6 requires entering Condition A when one control bank is inserted beyond the insertion limit and immovable. ITS 3.1.6 Required Action A.1 requires an immediate verification that the control bank is inserted less than or equal to 18 steps below the insertion limit as measured by the group step counter demand position indicators. ITS 3.1.5 Required Action A.2 requires an immediate verification that each control and shutdown rod is within the limits of LCO 3.1.4. ITS 3.1.5 Required Action A.3 requires an immediate verification that each shutdown bank is within the insertion limits of LCO 3.1.5. ITS 3.1.5 Required Action A.4 requires an immediate verification that there are no reactor coolant system boron concentration activities. ITS 3.1.5 Required Action A.5 requires an immediate verification that there are no power level increases. ITS 3.1.6 Required Action A.6 requires verification that the SDM is within the limits specified in the COLR once per 12 hours and upon insertion of the controlling bank more than 5 steps from the initial position. ITS 3.1.6 Required Action A.7 requires the restoration of the shutdown banks to within limits in 72 hours. This addition is acceptable because it reflects the current licensing basis. Furthermore, ISTS 3.1.6 Condition A (ITS 3.1.6 Condition B) was modified to state it is applicable for reasons other than Condition A, consistent with current licensing. This change was approved in License Amendment 215 for Unit 1 and License Amendment 205 for Unit 2 (ADAMS Accession No. ML013330266). Additionally, due to the addition of ITS 3.1.6 ACTION A, the subsequent ACTIONS (ISTS 3.1.5 ACTIONS A, B, and C) were renumbered.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.1.6.2 and SR 3.1.6.3 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in ~~all~~ the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate. 9

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements. 1

The control bank insertion limits are specified in the COLR. ~~An example is provided for information only in Figure B 3.1.6-1.~~ The control banks are required to be at or above the insertion limit lines. 2

~~Figure B 3.1.6-1 also indicates how the control banks are moved in an overlap pattern.~~ Overlap is the distance travelled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal~~, will be at 118 steps for a fully withdrawn position of 231 steps~~ is shown on the COLR Figure. The fully withdrawn position is defined in the COLR. 2 1

BASES

BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

    1. Specified acceptable fuel design limits or

    2. Reactor Coolant System pressure boundary integrity and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and control bank insertion limits affect safety analys~~i~~s involving core reactivity and power distributions (Ref. 3). e 9

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. ~~4~~ 3). 1

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may ~~also~~ indicate the maximum ejected RCCA worth could be equal to the limiting value in the fuel cycle~~s~~ that ~~have~~ has sufficiently high ejected RCCA worths. 9

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. ~~5~~ 3). 1

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analys~~i~~s. e 9

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with $k_{eff} < 1.0$, MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. 3

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

BASES

ACTIONS

INSERT 1 (5)

~~A~~B.1.1, ~~A~~B.1.2, ~~A~~B.2, ~~B~~C.1.1, ~~B~~C.1.2, and ~~B~~C.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap~~s~~ limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. (4)

~~C~~D.1

If Required Actions ( ) of Condition A, B, or C are not met ~~A.1 and A.2, or B.1 and B.2 cannot be completed~~ within the associated Completion Times, the plant must be brought to at least MODE 2 with $k_{eff} < 1.0$, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. (5) (9)

4 **INSERT 1**

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one control bank is inserted beyond the insertion limit and is immovable due to malfunctions in the rod control system, 72 hours are provided to restore the control banks to within limits. Additionally, immediate verification is required to prove that the control bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual rod alignment limits of LCOs 3.1.4 and 3.1.5 are met, there are no reactor coolant system boron concentration dilution activities, and there are no power level increases taking place. Furthermore, a verification of SDM is required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.5 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted bank. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Conditions are not met, Condition D must be applied.

The Condition is modified by a Note that specifies it only applies to control banks inserted beyond the insertion limit that are not controlling banks. A controlling bank is defined as a control bank that is less than fully withdrawn as defined in the COLR, with the exception of fully withdrawn banks that have been inserted for the performance of SR 3.1.4.2 (rod freedom of movement Surveillance).

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

~~[~~ Verification of the control bank insertion limits ~~at a Frequency of 12 hours~~ is sufficient to detect control banks that may be approaching the insertion limits ~~since, normally, very little rod motion occurs in 12 hours~~. 6

~~OR~~ 6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 7 6

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. ~~[A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.~~ 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~--]~~ 6

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.

2. 10 CFR 50.46.

3. U FSAR, Chapter ~~[~~15~~]~~. 1 8

~~4. FSAR, Chapter [15].~~ 1

~~5. FSAR, Chapter [15].~~

2

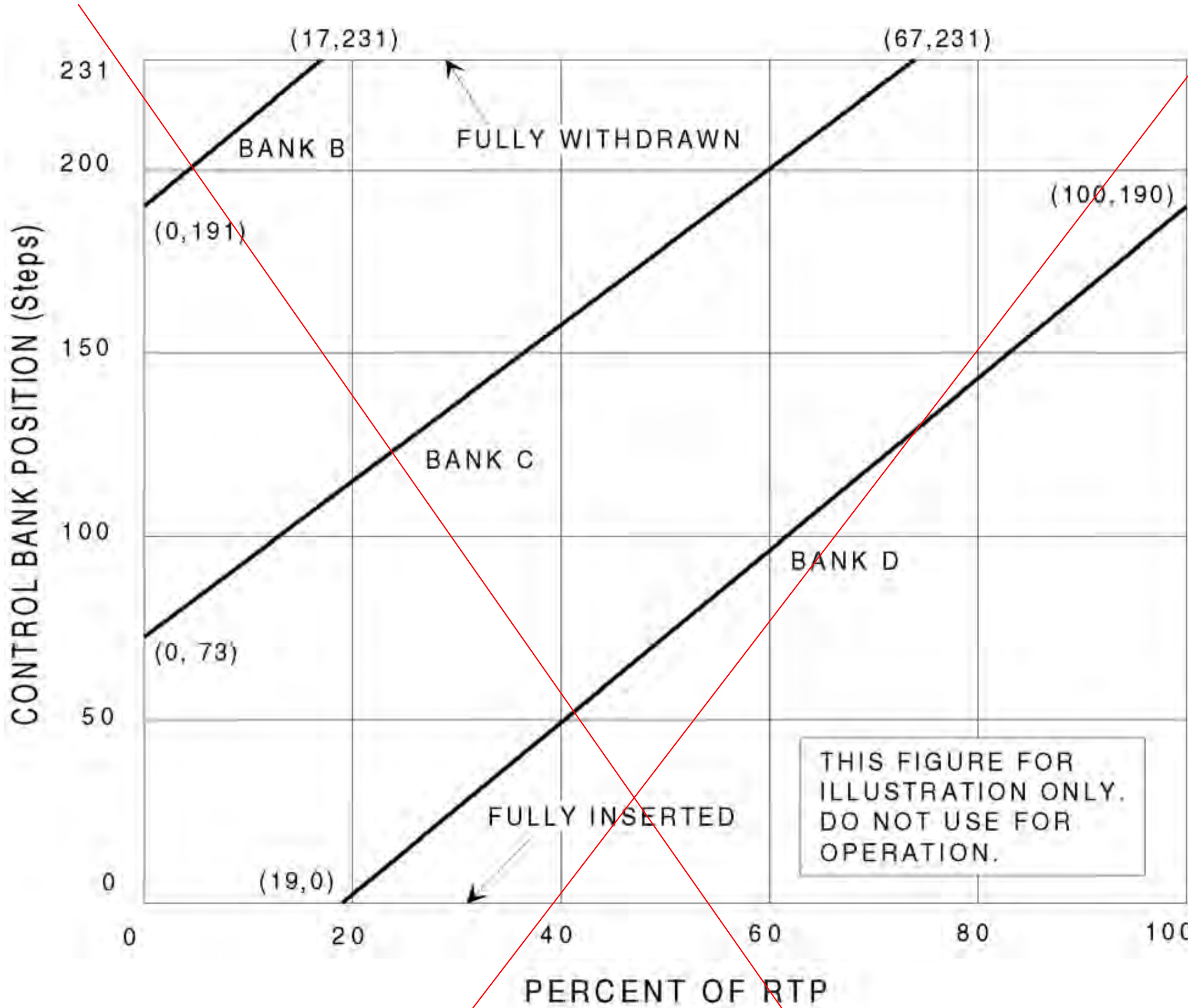

Figure B 3.1.6 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in ~~all~~ the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate. 9

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ Each unit has four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements. 1

The control bank insertion limits are specified in the COLR. ~~An example is provided for information only in Figure B 3.1.6-1.~~ The control banks are required to be at or above the insertion limit lines. 2

~~Figure B 3.1.6-1 also indicates how the control banks are moved in an overlap pattern.~~ Overlap is the distance travelled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal~~, will be at 118 steps for a fully withdrawn position of 231 steps~~ is shown on the COLR Figure. The fully withdrawn position is defined in the COLR. 2 1

BASES

BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

    1. Specified acceptable fuel design limits or

    2. Reactor Coolant System pressure boundary integrity and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and control bank insertion limits affect safety analys~~i~~es involving core reactivity and power distributions (Ref. 3). 9

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. ~~4~~3). 1

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may ~~also~~ indicate the maximum ejected RCCA worth could be equal to the limiting value in the fuel cycle~~s~~ that ~~have~~ has sufficiently high ejected RCCA worths. 9

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. ~~5~~3). 1

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analys~~i~~es. 9

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with $k_{eff} < 1.0$, MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. 3

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

BASES

ACTIONS

INSERT 1 5

~~A~~ B.1.1, ~~A~~ B.1.2, ~~A~~ B.2, ~~B~~ C.1.1, ~~B~~ C.1.2, and ~~B~~ C.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap~~s~~ limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. 4

~~C~~ D.1

If Required Actions ( ) of Condition A, B, or C are not met ~~A.1 and A.2, or B.1 and B.2 cannot be completed~~ within the associated Completion Times, the plant must be brought to at least MODE 2 with k_{eff} < 1.0, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. 5 9

4 **INSERT 1**

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one control bank is inserted beyond the insertion limit and is immovable due to malfunctions in the rod control system, 72 hours are provided to restore the control banks to within limits. Additionally, immediate verification is required to prove that the control bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual rod alignment limits of LCOs 3.1.4 and 3.1.5 are met, there are no reactor coolant system boron concentration dilution activities, and there are no power level increases taking place. Furthermore, a verification of SDM is required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.5 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted bank. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Conditions are not met, Condition D must be applied.

The Condition is modified by a Note that specifies it only applies to control banks inserted beyond the insertion limit that are not controlling banks. A controlling bank is defined as a control bank that is less than fully withdrawn as defined in the COLR, with the exception of fully withdrawn banks that have been inserted for the performance of SR 3.1.4.2 (rod freedom of movement Surveillance).

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

~~[~~ Verification of the control bank insertion limits ~~at a Frequency of 12 hours~~ is sufficient to detect control banks that may be approaching the insertion limits ~~since, normally, very little rod motion occurs in 12 hours~~. 6

~~OR~~ 6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~--]~~ 6

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. ~~[A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.~~ 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~---]~~ 6

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
2. 10 CFR 50.46.
3. U FSAR, Chapter ~~[~~15~~]~~. 1 8
4. ~~FSAR, Chapter [15].~~ 1
5. ~~FSAR, Chapter [15].~~

2

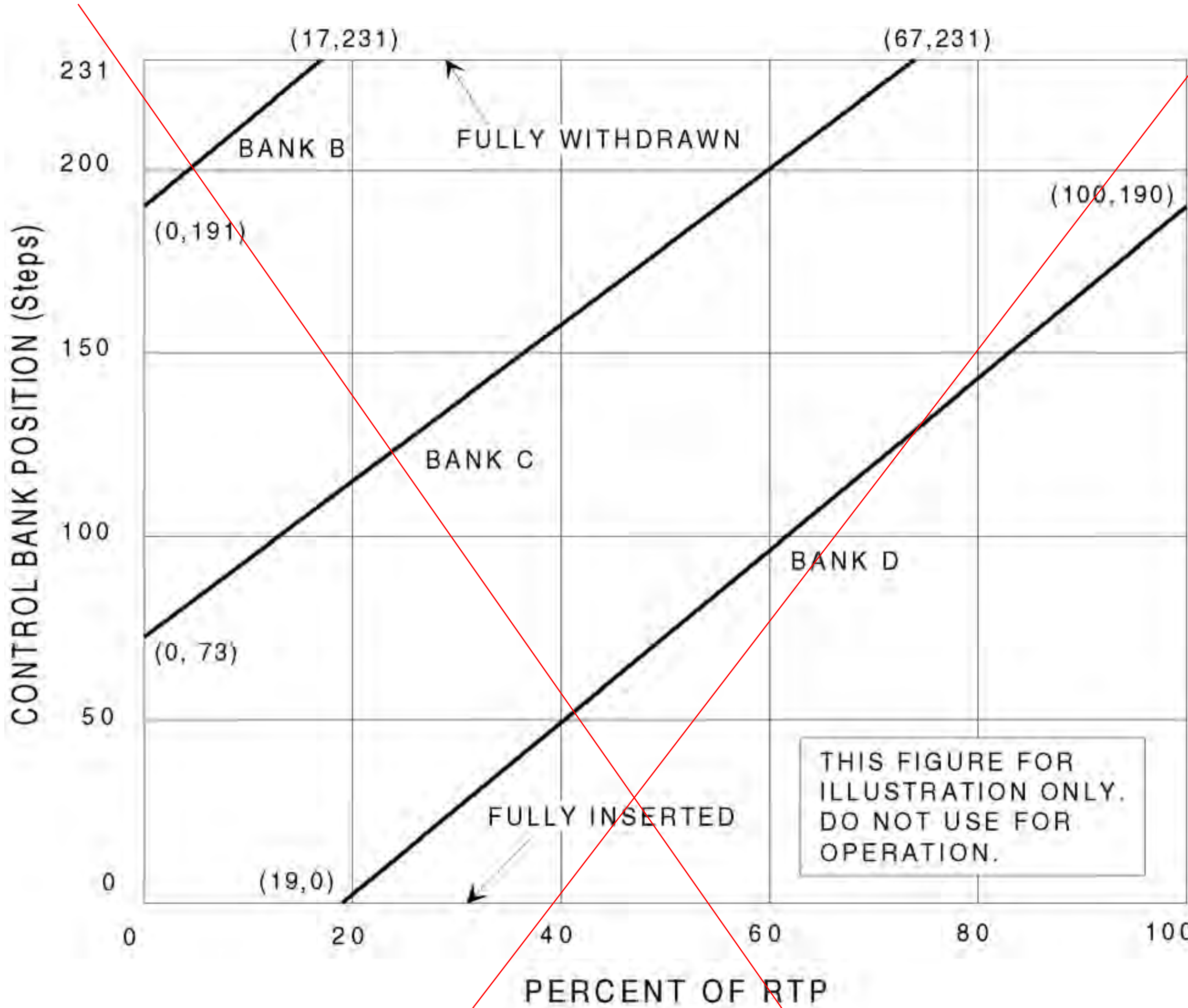

Figure B 3.1.6 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6 BASES, CONTROL BANK INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS 3.1.6 contains Figure B 3.1.6-1 and states that it is an example provided for information only. ITS 3.1.6 does not include Figure B 3.1.6-1. The control bank insertion limits for Sequoyah Nuclear Plant (SQN) are located in the COLR. Therefore, ISTS Figure B 3.1.6-1 and the references to the ISTS Figure B 3.1.6-1 have been deleted.

3. Changes are made to be consistent with the Specification.

4. Typographical/grammatical error corrected.

5. Changes are made to be consistent with changes made to the Specification.

6. ISTS SR 3.1.6.2 and SR 3.1.6.3 Bases provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

7. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

8. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

9. Editorial changes made for enhanced clarity/consistency.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS 3.1.7, ROD POSITION INDICATION

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

A01

ITS ITS 3.1.7

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.1.7 3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE ~~and capable of determining the control rod positions within ± 12 steps.~~ LA01

Applicability APPLICABILITY: MODES 1 and 2.

ACTION: L01

Add proposed ACTIONS Note 1

ACTION A

a. With a maximum of one rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 12 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

Condition C

MEH-004

2.* a) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours and once every 31 days thereafter and within 8 hours if rod control system parameters indicate unintended movement, and

b) Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable position indicator within 16 hours and once per 8 hours thereafter, and

c) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours if the rod with an inoperable position indicator is moved greater than 12 steps and prior to increasing THERMAL POWER above 50% RATED THERMAL POWER and within 8 hours of reaching 100% RATED THERMAL POWER, or

Required Action C.2

3. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

M01

Add proposed ACTION ~~D~~ E

ACTION B

b. With more than one rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 12 hours, and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and

Condition C

MEH-004

Required Action A.2 Note

* Rod position monitoring by Actions 2.a), 2.b), and 2.c) may only be applied to one inoperable rod position indicator ~~and shall only be allowed: (1) until the end of the current cycle, or (2) until an entry into MODE 5 of sufficient duration, whichever occurs first, when the repair of the inoperable rod position indication can safely be performed. Actions 2.a), 2.b), and 2.c) shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable rod position indication could have safely been performed.~~

Add proposed ACTIONS Note 2 A02

A01

ITS ITS 3.1.7

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - OPERATING

ACTION B

2. Place the control rods under manual control, and monitor and record Reactor Coolant System average temperature (T_{avg}) at least once per hour, and

3. Restore the rod position indicators to OPERABLE status within 24 hours such that a maximum of one rod position indicator per bank is inoperable, or

ACTION ~~D~~ E

Add Proposed Required Action C.2 MEH-004 L02

4. Be in HOT STANDBY within the next 6 hours.

MEH-004

c. With a maximum of one demand position indicator per bank inoperable either:

ACTION ~~C~~ D

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 12 hours, or

~~2.~~ Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

Add proposed ACTION ~~D~~ E M01

SURVEILLANCE REQUIREMENTS

~~4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours~~.

Add proposed SR 3.1.7.1 M02

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A01

ITS ITS 3.1.7

~~REACTIVITY CONTROL SYSTEMS~~

~~POSITION INDICATION SYSTEM - SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.3 This specification is deleted.~~

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A01

ITS

ITS 3.1.7

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.1.7

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE ~~and capable of determining the control rod positions within ± 12 steps~~.

LA01

Applicability

APPLICABILITY: Modes 1 and 2.

ACTION:

L01

Add proposed ACTIONS Note 1

a. With a maximum of one rod position indicator per bank inoperable either:

Condition C

MEH-004

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 12 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

ACTION A

2.* a) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours and once every 31 days thereafter and within 8 hours if rod control system parameters indicate unintended movement, and

b) Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable position indicator within 16 hours and once per 8 hours thereafter, and

c) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours if the rod with an inoperable position indicator is moved greater than 12 steps and prior to increasing THERMAL POWER above 50% RATED THERMAL POWER and within 8 hours of reaching 100% RATED THERMAL POWER, or

Required Action C.2

3. Reduce THERMAL POWER TO less than 50% of RATED THERMAL POWER within 8 hours.

M01

Add proposed ACTION ~~D~~ E

MEH-004

b. With more than one rod position indicator per bank inoperable either:

Condition C

ACTION B

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 12 hours, and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and

Required Action A.2 Note

* Rod position monitoring by Actions 2.a), 2.b), and 2.c) may only be applied to one inoperable rod position indicator ~~and shall only be allowed: (1) until the end of the current cycle, or (2) until an entry into MODE 5 of sufficient duration, whichever occurs first, when the repair of the inoperable rod position indication can safely be performed. Actions 2.a), 2.b), and 2.c) shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable rod position indication could have safely been performed.~~

Add proposed ACTIONS Note 2

A02

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A01

ITS ITS 3.1.7

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

ACTION B

2. Place the control rods under manual control, and monitor and record Reactor Coolant System average temperature (T_{avg}) at least once per hour, and

3. Restore the rod position indicators to OPERABLE status within 24 hours such that a maximum of one rod position indicator per bank is inoperable, or

ACTION ~~D~~ E

4. Be in HOT STANDBY within the next 6 hours.

Add Proposed Required Action C.2

MEH-004

L02

ACTION ~~C~~ D

c. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 12 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

Add proposed ACTION ~~D~~ E

M01

SURVEILLANCE REQUIRMENTS

~~4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.~~

Add proposed SR 3.1.7.1

M02

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~~REACTIVITY CONTROL SYSTEMS~~

~~POSITION INDICATION SYSTEM-SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.3 This specification is deleted.~~

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DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.1.3.2 Note * applies to Actions 2.a, 2.b, and 2.c and may be only applied to one inoperable rod position indicator. In this condition, the inoperable rod position indicator shall only be allowed until either the end of the current cycle, or until an entry into MODE 5 of sufficient duration, whichever occurs first, when the repair of the inoperable rod position indication can safely be performed. Actions 2.a, 2.b, and 2.c shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable rod position indication could have safely been performed. ITS 3.1.7 ACTIONS Note 2 states that LCO 3.0.4.a and b are not applicable for Required Actions A.2.1 and A.2.2 following startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication. This changes the CTS by rewording the allowance for one rod position indicator inoperable to be consistent with ITS terminology.

This change is designated as an administrative change since the change does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.2 ACTION a and c do not contain an ACTION to follow if the provided ACTIONS cannot be met. Therefore, CTS 3.0.3 would be entered, which would allow 1 hour to initiate a shutdown and 7 hours to be in HOT STANDBY. E ITS 3.1.7 ACTION ~~D~~ requires if the Required Actions and associated Completion Time S ~~of ACTION A or C~~ are not met, to be in MODE 3 within 6 hours. This changes the CTS by eliminating the one hour to initiate a shutdown and consequently allows one hour less for the unit to be in MODE 3. MEH-004

This change is acceptable because it provides an appropriate compensatory measure for the described conditions. If any Required Action and associated Completion Time cannot be met, the unit must be placed in a MODE in which the LCO does not apply. The LCO is applicable in MODES 1 and 2. Requiring a shutdown to MODE 3 is appropriate in this condition. The one hour allowed by CTS 3.0.3 to prepare for a shutdown is not needed because the operators have had time to prepare for the shutdown while attempting to follow the Required Actions and associated Completion Times. This change is designated as more restrictive because it allows less time to shutdown than is allowed in the CTS.

DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

M02 CTS 4.1.3.2 requires that each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours. ITS 3.1.7 does not contain this requirement because it is duplicative of CTS 4.1.3.1.1 (ITS SR 3.1.4.1). A new Surveillance has been added (ITS SR 3.1.7.1) to verify each RPI agrees within 12 steps of the group demand position for the full indicated range of rod travel, once prior to criticality after each removal of the reactor head. This changes the CTS by adding a new Surveillance Requirement.

The purpose of ITS SR 3.1.7.1 is to provide additional assurance that the rod position indication system is operating correctly. This change is acceptable because it provides additional assurance that the rod position indication channels are OPERABLE. This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.1.3.2 requires the shutdown and control rod position indication system and the demand position indication system to be OPERABLE and capable of determining the control rod positions within ± 12 steps. ITS LCO 3.1.7 requires the analog Rod Position Indication System and the Demand Position Indication System to be OPERABLE but the details of what constitutes an OPERABLE system are moved to the Bases. This changes the CTS by removing the details of what constitutes an OPERABLE system to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement that the Rod Position Indication System and Demand Position Indication System be OPERABLE. The details on the capability requirements of the systems do not need to appear in the specification in order for the requirement to apply. Additionally, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.2 ACTION a covers the inoperability for a maximum of one rod position indicator per bank. CTS 3.1.3.2 ACTION b covers the inoperability for more than one rod position indicator per bank. CTS 3.1.3.2 ACTION c covers the inoperability for a maximum of one demand position indicator per bank. ITS 3.1.7 ACTIONS are modified by Note 1 that states "Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator." ITS 3.1.7 ACTION A covers inoperability for one rod position indicator per bank. ITS 3.1.7 ACTION B covers inoperability for more than one rod position indicator per bank. ITS 3.1.7 ACTION ~~C~~ D covers inoperability for one demand position indicator bank for one or more banks. This changes the CTS by allowing separate Condition entry for each inoperable rod position indicator and each demand position indicator. MEH-004

The purpose of CTS 3.1.3.2 ACTION a is to provide compensatory actions for a maximum of one rod position indicator per bank. The purpose of CTS 3.1.3.2 ACTION b is to provide compensatory actions for more than one rod position indicator per bank. The purpose of CTS 3.1.3.2 ACTION c is to provide compensatory actions for one demand position indicator per bank. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. This change will allow separate Condition entry for each inoperable rod position indicator and each inoperable demand position indicator while the CTS does not. The ITS will allow each inoperable rod position indicator or each inoperable demand position indicator to be tracked separately. This change is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for inoperable position indication. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

MEH-004

L02

DOC L02 MEH-004

(Category 4 – Relaxation of Required Action) The CTS 3.1.3.2 ACTION for, “more than one rod position indicator per bank inoperable” requires the performance of ACTION b.1 and b.2 and b.3 or b.4. If CTS Actions b.1, b.2, and b.3 are not performed, then CTS 3.1.3.2 Action b.4 requires placing the unit in HOT STANDBY within the next 6 hours. CTS 3.1.3.2 Action b.1 requires, in part, determining the position of the non-indicating rod(s) indirectly by the movable incore detectors immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod’s position. ITS 3.1.7 Condition C for, “one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod’s position,” requires performance of Required Action C.1 or C.2. ITS 3.1.7 Required Action C.1 requires verifying the position of the rods with inoperable position indicators indirectly by using the movable incore detectors with a Completion Time of immediately. ITS 3.1.7 Required Action C.2 requires reducing THERMAL POWER to < 50% RTP. This changes the CTS by allowing a reduction in THERMAL POWER as an alternative to verifying the position of the rods with inoperable position indicators and placing the unit in HOT STANDBY within the next 6 hours.

The purpose of the CTS 3.1.3.2 ACTION to “verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors” is to ensure a misaligned rod does not go undetected and cause a power imbalance in the core. This change is acceptable because, if the rod positions have not been determined, THERMAL POWER must be reduced to < 50% RTP to avoid undesirable power distributions that could result from continued operation at ≥ 50% RTP when one or more rods are misaligned by more than 24 steps. This change is designated as less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

3.1.3.2 LCO 3.1.7 The ~~[Digital]~~ Rod Position Indication ~~([D]RPI)~~ System and the Demand Position Indication System shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

S

---NOTE--- 5

1. Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a MEH-004 A. One ~~[D]RPI~~ rod position indicator per ~~group~~ bank inoperable ~~for one or more groups~~.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. INSERT 3 OR A.~~2~~ 3 Reduce THERMAL POWER to ~~≤~~ < 50% RTP.	Once per ~~8~~ 12 hours 1 4 ~~INSERT 2~~ ~~2~~ 3 8 hours 3 4
ACTION b B. More than one ~~[D]RPI~~ rod position indicator per ~~group~~ bank inoperable.	B.1 Place the control rods under manual control. AND B.2 Monitor and record Reactor Coolant System T_{avg}. AND	Immediately 1 4 Once per 1 hour

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4 **INSERT 1**

3.1.3.2 Note* 2. LCO 3.0.4.a and b are not applicable for Required Actions A.2.1 and A.2.2 following a startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

~~2~~ ~~**INSERT 2**~~

~~AND~~ MEH-004

~~Action a.1~~ ~~Immediately after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position~~

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3

INSERT 3

OR	
--------------------NOTE------------------- Required Actions A.2.1 and A.2.2 may only be applied to one inoperable rod position indicator. ---	
A.2.1 Verify position of the rod with inoperable position indicator indirectly by using movable incore detectors.	8 hours AND Once per 31 days thereafter AND 8 hours if Rod Control System parameters indicate unintended movement AND 8 hours if the rod with an inoperable position indicator is moved greater than 12 steps AND Prior to increasing THERMAL POWER above 50% RTP AND 8 hours after reaching 100% RTP
AND	

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3 **INSERT 3 (Continued)**

A.2.2	Review the parameters of the Rod Control System for indications of unintended rod movement for the rod with the inoperable position indicator.	16 hours AND Once per 8 hours thereafter

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION b		B.3 Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per ~~8~~ 12 hours ~~INSERT 4~~	4 ~~2~~ MEH-004
		AND		
STET		B.4 Restore inoperable position indicators to OPERABLE status such that a maximum of one ~~[D]RPI~~ rod position indicator per ~~group~~ bank is inoperable.	24 hours Immediately	1 4
ACTION a.1 ACTION b.1 ACTION a.3 DOC L02	~~C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~	~~C.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.~~	~~[4] hours~~	1 MEH-004 ~~2~~
		~~OR~~		
		~~C.2 Reduce THERMAL POWER to ≤ 50% RTP.~~ <	~~8 hours~~	4
DOC L01 ACTION c STET	~~D~~ C. One demand position indicator per bank inoperable for one or more banks.	~~D~~ C.1.1 Verify by administrative means all ~~[D]RPIs~~ rod position indicators for the affected banks are OPERABLE.	Once per ~~8~~ 12 hours	~~2~~ 4 1 MEH-004
		AND		
		~~D~~ C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per ~~8~~ 12 hours	~~2~~ 4
		OR		

3.1.7

2 **~~INSERT 4~~**

MEH-004

~~AND~~

~~Action b.1~~ ~~Immediately after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position~~

~~Insert Page 3.1.7-2~~

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION c		~~D~~ ~~C~~.2	Reduce THERMAL POWER to ~~≤~~ < 50% RTP.	8 hours	~~2~~ 4 MEH-004
ACTION b.4, DOC M02	~~E~~ ~~D~~. Required Action and associated Completion Time not met.	~~E~~ ~~D~~.1	Be in MODE 3.	6 hours	~~2~~

STET

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.3.2	SR 3.1.7.1	Verify each ~~[D]RPI~~ rod position indicator agrees within ~~[~~12~~]~~ steps of the group demand position ~~for the [full indicated range]~~ at 20 and 215 steps of rod travel.	Once prior to criticality after each removal of the reactor head	1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

3.1.3.2 LCO 3.1.7 The ~~[Digital]~~ Rod Position Indication ~~([D]RPI)~~ System and the Demand Position Indication System shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

S

---NOTE--- 5

1. Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a A. One ~~[D]RPI~~ rod position indicator per ~~group~~ bank inoperable ~~for one or more groups~~.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. INSERT 3 OR A.~~2~~ 3 Reduce THERMAL POWER to ~~≤~~ < 50% RTP.	Once per ~~8~~ 12 hours 1 4 ~~INSERT 2~~ 2 MEH-004 3 8 hours 3 4
ACTION b B. More than one ~~[D]RPI~~ rod position indicator per ~~group~~ bank inoperable.	B.1 Place the control rods under manual control. AND B.2 Monitor and record Reactor Coolant System T_{avg}. AND	Immediately 1 4 Once per 1 hour

4 **INSERT 1**

3.1.3.2 Note* 2. LCO 3.0.4.a and b are not applicable for Required Actions A.2.1 and A.2.2 following a startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

~~2~~ **~~INSERT 2~~**

~~AND~~ MEH-004

~~Action a.1~~ ~~Immediately after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position~~

3 **INSERT 3**

OR --------------------NOTE------------------- Required Actions A.2.1 and A.2.2 may only be applied to one inoperable rod position indicator. --	
A.2.1 Verify position of the rod with inoperable position indicator indirectly by using movable incore detectors.	8 hours AND Once per 31 days thereafter AND 8 hours if Rod Control System parameters indicate unintended movement AND 8 hours if the rod with an inoperable position indicator is moved greater than 12 steps AND Prior to increasing THERMAL POWER above 50% RTP AND 8 hours after reaching 100% RTP
AND	

3 **INSERT 3 (Continued)**

A.2.2	Review the parameters of the Rod Control System for indications of unintended rod movement for the rod with the inoperable position indicator.	16 hours <u>AND</u> Once per 8 hours thereafter

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	B.3 Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per ~~8~~ 12 hours ~~INSERT 4~~
	AND	
	B.4 Restore inoperable position indicators to OPERABLE status such that a maximum of one ~~[D]RPI~~ rod position indicator per ~~group~~ bank is inoperable.	24 hours
STET ACTION a.1 ACTION b.1 ACTION a.3 DOC L02 ~~C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~	~~C.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.~~	Immediately ~~[4] hours~~
	~~OR~~	
	~~C.2 Reduce THERMAL POWER to ≤ 50% RTP.~~ <	~~8 hours~~
DOC L01 ACTION c ~~D~~ C. One demand position indicator per bank inoperable for one or more banks.	~~D~~ C.1.1 Verify by administrative means all ~~[D]RPIs~~ rod position indicators for the affected banks are OPERABLE.	Once per ~~8~~ 12 hours
STET	AND	
	~~D~~ C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per ~~8~~ 12 hours
	OR	

4 ~~2~~ MEH-004 1 4 1 MEH-004 ~~2~~ 4 ~~2~~ 4 1 ~~2~~ 4

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2 ~~**INSERT 4**~~

~~AND~~

~~Action b.1~~ ~~Immediately after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position~~ MEH-004

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION c	STET	~~D~~ ~~C~~.2	Reduce THERMAL POWER to ~~≤~~ < 50% RTP.	8 hours	~~2~~ 4 MEH-004
ACTION b.4, DOC M02	~~E~~ ~~D~~. Required Action and associated Completion Time not met.	~~E~~ ~~D~~.1	Be in MODE 3.	6 hours	~~2~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.3.2	SR 3.1.7.1	Verify each ~~[D]RPI~~ rod position indicator agrees within ~~[~~12~~]~~ steps of the group demand position ~~for the [full indicated range]~~ at 20 and 215 steps of rod travel.	Once prior to criticality after each removal of the reactor head	1

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, ROD POSITION INDICATION

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ~~ISTS 3.1.7 ACTION C has been deleted and a new conditional Completion time has been added to Required Action A.1 and B.3. The new completion time ensures that SQN current licensing basis is maintained, in that a verification of the position indicator is still being performed immediately after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position. Additionally, ISTS 3.1.7 ACTIONS D and E has been changed to ITS 3.1.7 ACTIONS C and D, respectively, because of this deletion.~~ MEH-004

Not Used

3. ISTS 3.1.7 ACTION A provides compensatory actions for when one rod position indicator is inoperable. ITS 3.1.7 provides an additional Required Action that can be taken when one rod position indicator is inoperable. The new Required Action allows the use of an alternate means other than the movable incore detectors to monitor the position of a control or shutdown rod when the analog rod position indication system is inoperable. This change reflects a current licensing basis that was approved by the NRC in Amendment 315 for Unit 1 and Amendment 304 for Unit 2 (ADAMS Accession No. ML063120575). Additionally ISTS 3.1.7 Required Action A.2 has been renumbered as ITS 3.1.7 Required Action A.3.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. Editorial changes made for enhanced clarity/consistency.

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control and shutdown rod position indicators to determine ~~control~~ rod positions and thereby ensure compliance with the control rod alignment and insertion limits. 1

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a ~~control~~ rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, resulting from ~~due to~~ the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. 5

Limits on control rod alignment and OPERABILITY have been established, and ~~all~~ rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. 5

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the ~~Bank~~ Demand Position Indication System (commonly called group step counters) and the ~~[Digital]~~ Rod Position Indication ~~([D]RPI)~~ System. 1 2

BASES

BACKGROUND (continued)

The ~~Bank~~ Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group ~~all~~ receive the same signal to move and should, therefore, ~~all~~ be at the same position indicated by the group step counter for that group. The ~~Bank~~ Demand Position Indication System is considered highly precise (± 1 step or ± ~~6~~ 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. 1 5 5 1 1

The ~~[D]RPI~~ Rod Position Indication System provides ~~a highly accurate~~ an indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube ~~with a center to center distance of 3.75 inches, which is 6 steps~~. ~~To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the [D]RPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the [D]RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches).~~ INSERT 1 ~~With an indicated deviation of 12 steps between the group step counter and [D]RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.~~ 2 1 1

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication ~~is~~ are that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). ~~Control~~ rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions. 5 5

The ~~control~~ rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The ~~control~~ rod position indicators monitor ~~control~~ rod position, which is an initial condition of the accident. 5

1 **INSERT 1**

A deviation of ± 12 steps between the group step counter and a rod position indication is based on normal Rod Position Indication System indication accuracy of ± 5% span with a maximum uncertainty of 10% span between the group step counter and the rod position indication.

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BASES

LCO

LCO 3.1.7 specifies that one ~~[D]RPI~~ Rod Position Indication System ~~and one Bank Demand Position Indication System~~ shall be OPERABLE for each ~~control~~ rod. INSERT 2 For the ~~control~~ rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following: 2 1 5

a. The ~~[D]RPI~~ Rod Position Indication System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits," 2

b. For the ~~[D]RPI~~ Rod Position Indication System there are no failed coils, and 2

c. The ~~Bank~~ Demand Position Indication System has been calibrated either in the fully inserted position or ~~to the [D]RPI System~~ INSERT 3. 1 4 1

The 12 step agreement limit between the ~~Bank~~ Demand Position Indication System and the ~~[D]RPI~~ Rod Position Indication System indicates that the ~~Bank~~ Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position. 2 1

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements ~~on~~ of the ~~[D]RPI~~ Rod Position Indication and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. 5 2

1 **INSERT 2**

Additionally, one Demand Position Indication System shall be OPERABLE for each group within a bank.

1 **INSERT 3**

a check is performed between the two step counters in the same bank. Shutdown Banks C and D each contain a single group. Therefore, validation of movement for Shutdown Banks C and D can only be performed with a comparison of the single group to the corresponding RPI movement.

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BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

INSERT 4 4

A.1

Rod Position Indication bank

When one ~~[D]RPI~~ channel per ~~group~~ fails, the position of the rod may ~~still~~ be determined indirectly by use of the movable incore detectors. ~~The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved.~~ Based on experience, normal power operation does not require excessive movement of banks. ~~If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required.~~ Therefore, verification of RCCA position within the Completion Time of ~~8~~ 12 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. 2 1 5 3 MEH-004 STET 4 4

SII

INSERT 5 4

A.~~2~~ 3 4

Reduction of THERMAL POWER to ~~≤~~ < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3). 4

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ~~≤~~ < 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above. 4

B.1, B.2, B.3, and B.4

Rod Position Indication bank fails

When more than one ~~[D]RPI~~ per ~~group fail~~, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via 2 1 5

4 **INSERT 4**

A second Note has been added to provide clarification that LCO 3.0.4.a and LCO 3.0.4.c are not applicable for Required Action A.2.1 and A.2.2 following startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

4 **INSERT 5** SII

~~If one or more rods have been significantly moved (in excess of 24 steps in one direction, since the position was last determined), Required Action A.1 is still appropriate, but actions must be initiated immediately to begin verifying that the rod is still properly positioned, relative to their group positions. In this Required Action, the Completion Time only begins on discovery that both:~~ MEH-004

~~a. One rod position indication per bank is inoperable, and~~

~~b. A rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~

~~If at any time during the existence of Condition A (one RPI per bank inoperable), a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position, this Completion Time begins to be tracked~~.

A.2.1, and A.2.2

When one RPI channel per bank fails, the position of the rod may still be determined indirectly by use of the movable incore detectors and reviewing the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication. Therefore, verification of RCCA position within 8 hours and every 31days thereafter is adequate for allowing continued full power operation as long as a review of the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication is performed within 16 hours and every 8 hours thereafter. Furthermore, if the rod control system parameters indicate unintended movement or if the rod with an inoperable position indicator is moved greater than 12 steps, then the verification of the RCCA position must be performed within 8 hours. As long as these compensatory actions are met, reactor operation can then continue until the end of the current cycle or until an entry into MODE 5 of sufficient duration that the repair of the inoperable rod position indication can safely be performed.

Required Actions A.2.1,and A.2.2 are modified by a Note directing that these Required Actions may only be applied to one inoperable rod position indicator.

BASES

ACTIONS (continued)

movable incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

helps

Monitoring and recording reactor coolant T_{avg} ~~help~~ assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions. 5

The position of the rods may be determined indirectly by use of the movable incore detectors. ~~The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved.~~ Verification of control rod position once per ~~8~~ 12 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the ~~[D]RPI~~ Rod Position Indication system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication. 3 4 2 5

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved~~, the Required Action of C.1 or C.2 below is required~~. ~~INSERT 6~~ STET 4

MEH-004

~~C.1 and C.2~~

~~These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2, [or B.1, as applicable] are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned, relative to their group positions.~~ 4

~~If, within [4] hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of [4] hours provides an acceptable period of time to verify the rod positions.~~ < ≥ 4

4 ~~**INSERT 6**~~

~~(in excess of 24 steps in one direction, since the position was last determined), Required Action B.3 is still appropriate, but action must be initiated immediately to begin verifying that the rod is properly positioned, relative to its bank position. In this Required Action, the Completion Time only begins on discovery that both:~~ MEH-004

~~a. More than one RPI per bank is inoperable; and~~

~~b. A rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~

~~If at any time during the existence of Condition B (more than one RPI per bank inoperable), a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position, this Completion Time begins to be tracked.~~

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BASES

ACTIONS (continued)

C
~~D~~.1.1 and ~~D~~.1.2 4

STET MEH-004

Rod Position Indication
With one demand position indicator per bank inoperable, the rod positions can be determined by the ~~[D]RPI~~ System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every ~~8~~ 12 hours is adequate. 2 4

C
~~D~~.2 4

STET

<
Reduction of THERMAL POWER to ~~≤~~ < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions ~~per Required Actions C.1.1 and C.1.2~~ or reduce power to ~~≤~~ < 50% RTP. 4 MEH-004 4 1 5

C.1 C.2

D
~~E~~.1 4

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Rod Position Indication
Verification that the ~~[D]RPI~~ agrees with the demand position within ~~[~~12~~]~~ steps ensures that the ~~[D]RPI~~ is operating correctly. ~~Since the [D]RPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~ INSERT 7 2 1

Rod Position Indication

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

1 **INSERT 7**

This verification will be performed at 20 steps and 215 steps of rod travel.

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BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.

Section 7.7.1

2. FSAR, ~~Chapter [15]~~. 1 2

U

3. FSAR, Chapter ~~[~~15~~]~~. 1 2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control and shutdown rod position indicators to determine ~~control~~ rod positions and thereby ensure compliance with the control rod alignment and insertion limits. 1

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a ~~control~~ rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, resulting from ~~due to~~ the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. 5

Limits on control rod alignment and OPERABILITY have been established, and ~~all~~ rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. 5

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the ~~Bank~~ Demand Position Indication System (commonly called group step counters) and the ~~[Digital]~~ Rod Position Indication ~~([D]RPI)~~ System. 1 2

BASES

BACKGROUND (continued)

The ~~Bank~~ Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group ~~all~~ receive the same signal to move and should, therefore, ~~all~~ be at the same position indicated by the group step counter for that group. The ~~Bank~~ Demand Position Indication System is considered highly precise (± 1 step or ± ~~☐~~ 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. 1 5 5 1 1

The ~~[D]RPI~~ Rod Position Indication System provides ~~a highly accurate~~ an indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube ~~with a center to center distance of 3.75 inches, which is 6 steps~~. ~~To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the [D]RPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the [D]RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches).~~ INSERT 1 ~~With an indicated deviation of 12 steps between the group step counter and [D]RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.~~ 2 1 1

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication ~~is~~ are that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). ~~Control~~ rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions. 5 5

The ~~control~~ rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The ~~control~~ rod position indicators monitor ~~control~~ rod position, which is an initial condition of the accident. 5

1 **INSERT 1**

A deviation of ± 12 steps between the group step counter and a rod position indication is based on normal Rod Position Indication System indication accuracy of ± 5% span with a maximum uncertainty of 10% span between the group step counter and the rod position indication.

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BASES

LCO

Rod Position Indication shall

LCO 3.1.7 specifies that one ~~[D]RPI~~ System ~~and one Bank Demand Position Indication System~~ be OPERABLE for each ~~control~~ rod. For the ~~control~~ rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following: 2 1 5

INSERT 2

Rod Position Indication

a. The ~~[D]RPI~~ System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits," 2

Rod Position Indication

b. For the ~~[D]RPI~~ System there are no failed coils, and 2

Position

c. The ~~Bank~~ Demand Indication System has been calibrated either in the fully inserted position or ~~to the [D]RPI System~~. 1 4 1

INSERT 3

Rod Position Indication

The 12 step agreement limit between the ~~Bank~~ Demand Position Indication System and the ~~[D]RPI~~ System indicates that the ~~Bank~~ Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position. 2 1

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicators channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

of Rod Position Indication

The requirements ~~on~~ the ~~[D]RPI~~ and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. 5 2

1 **INSERT 2**

Additionally, one Demand Position Indication System shall be OPERABLE for each group within a bank.

1 **INSERT 3**

a check is performed between the two step counters in the same bank. Shutdown Banks C and D each contain a single group. Therefore, validation of movement for Shutdown Banks C and D can only be performed with a comparison of the single group to the corresponding RPI movement.

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BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

INSERT 4 (4)

A.1

Rod Position Indication

bank

When one ~~[D]RPI~~ channel per ~~group~~ fails, the position of the rod may ~~still~~ be determined indirectly by use of the movable incore detectors. ~~The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved.~~ Based on experience, normal power operation does not require excessive movement of banks. ~~If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required.~~ Therefore, verification of RCCA position within the Completion Time of ~~8~~ 12 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. (2) (1) (5) (3) (4) (4)

STET

MEH-004

SII

INSERT 5 (4)

A.~~2~~ 3 (4)

Reduction of THERMAL POWER to ~~≤~~ < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3). (4)

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ~~≤~~ < 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above. (4)

B.1, B.2, B.3, and B.4

Rod Position Indication

bank

fails

When more than one ~~[D]RPI~~ per ~~group fail~~, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via (2) (1) (5)

4 **INSERT 4**

A second Note has been added to provide clarification that LCO 3.0.4.a and LCO 3.0.4.c are not applicable for Required Action A.2.1 and A.2.2 following startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

4 **INSERT 5** SII

~~If one or more rods have been significantly moved (in excess of 24 steps in one direction, since the position was last determined), Required Action A.1 is still appropriate, but actions must be initiated immediately to begin verifying that the rod is still properly positioned, relative to their group positions. In this Required Action, the Completion Time only begins on discovery that both:~~ MEH-004

~~a. One rod position indication per bank is inoperable, and~~

~~b. A rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~

~~If at any time during the existence of Condition A (one RPI per bank inoperable), a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position, this Completion Time begins to be tracked.~~

A.2.1, and A.2.2

When one RPI channel per bank fails, the position of the rod may still be determined indirectly by use of the movable incore detectors and reviewing the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication. Therefore, verification of RCCA position within 8 hours and every 31days thereafter is adequate for allowing continued full power operation as long as a review of the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication is performed within 16 hours and every 8 hours thereafter. Furthermore, if the rod control system parameters indicate unintended movement or if the rod with an inoperable position indicator is moved greater than 12 steps, then the verification of the RCCA position must be performed within 8 hours. As long as these compensatory actions are met, reactor operation can then continue until the end of the current cycle or until an entry into MODE 5 of sufficient duration that the repair of the inoperable rod position indication can safely be performed.

Required Actions A.2.1,and A.2.2 are modified by a Note directing that these Required Actions may only be applied to one inoperable rod position indicator.

BASES

ACTIONS (continued)

movable incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

helps

Monitoring and recording reactor coolant T_{avg} ~~help~~ assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions. 5

The position of the rods may be determined indirectly by use of the movable incore detectors. ~~The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved.~~ Verification of control rod position once per ~~8~~ 12 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the ~~[D]RPI~~ Rod Position Indication system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication. 3 4 2 5

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved~~, the Required Action of C.1 or C.2 below is required~~. STET ~~INSERT 6~~ 4

MEH-004

~~C.1 and C.2~~

~~These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2, [or B.1, as applicable] are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned, relative to their group positions.~~ 4

<

~~If, within [4] hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of [4] hours provides an acceptable period of time to verify the rod positions.~~ 4

≥

4 ~~**INSERT 6**~~

~~(in excess of 24 steps in one direction, since the position was last determined), Required Action B.3 is still appropriate, but action must be initiated immediately to begin verifying that the rod is properly positioned, relative to its bank position. In this Required Action, the Completion Time only begins on discovery that both:~~ MEH-004

~~a. More than one RPI per bank is inoperable; and~~

~~b. A rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position.~~

~~If at any time during the existence of Condition B (more than one RPI per bank inoperable), a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position, this Completion Time begins to be tracked.~~

BASES

ACTIONS (continued)

C
~~D~~.1.1 and ~~D~~.1.2 4

STET

MEH-004

Rod Position Indication

With one demand position indicator per bank inoperable, the rod positions can be determined by the ~~[D]RPI~~ System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every ~~8~~ 12 hours is adequate. 2 4

C
~~D~~.2 4

STET

<

Reduction of THERMAL POWER to ~~≤~~ < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions ~~per Required Actions C.1.1 and C.1.2~~ or reduce power to ~~≤~~ < 50% RTP. 4 MEH-004 4 1 5

C.1 C.2

D
~~E~~.1 4

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Rod Position Indication

Verification that the ~~[D]RPI~~ agrees with the demand position within ~~[~~12~~]~~ steps ensures that the ~~[D]RPI~~ is operating correctly. ~~Since the [D]RPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~ 2 1

Rod Position Indication

INSERT 7

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

1 **INSERT 7**

This verification will be performed at 20 steps and 215 steps of rod travel.

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BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.

U

Section 7.7.1

2. FSAR, ~~Chapter [15]~~. 1 2

3. FSAR, Chapter ~~[~~15~~]~~. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7 BASES, ROD POSITION INDICATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS 3.1.7 Required Action A.1 Bases contains a statement allowing an alternative method of satisfying Required Action A.1 by verifying that F_Q and $F^N_{\Delta H}$ are within the limits provided in the COLR, provided the nonindicating rods have not been moved. Additionally, ISTS 3.1.7 Required Action B.3 Bases also contains this statement. ITS 3.1.7 Required Action A.1 Bases and Required Action B.3 Bases do not contain this statement. The statement has been deleted because it allows an alternative method for satisfying Required Actions A.1 and B.3 that are not addressed in the Specification. Since the Technical Specification Bases are not allowed to modify the Technical Specifications, this statement has been deleted.

4. Changes are made to be consistent with changes made to the Specification.

5. Editorial changes made for enhanced clarity/consistency.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.7, ROD POSITION INDICATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

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

ITS

A01

ITS 3.1.8

3.1 REACTIVITY CONTROL SYSTEMS

~~SPECIAL TEST EXCEPTIONS~~

3/4.10.3 PHYSICS TESTS

Exceptions – MODE 2

A02

LIMITING CONDITION FOR OPERATION

INSERT 1

A03

LCO 3.1.8

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

~~b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels low trip setpoints are set at less than or equal to 25% of RATED THERMAL POWER, and~~

A04

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

Add proposed LCO 3.1.8.b

M01

Applicability

APPLICABILITY: MODE 2.

During PHYSICS TESTS initiated in

A05

ACTION:

Add proposed ACTION A

M01

ACTION B

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

ACTION C

ACTION D

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limits within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER ~~at least once per hour during PHYSICS TESTS~~.

In accordance with the Surveillance Frequency Control Program

LA01

SR 3.1.8.1

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL ~~FUNCTIONAL~~ TEST prior to initiating PHYSICS TESTS.

OPERATIONAL

M02

SR 3.1.8.2

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F ~~at least once per 30 minutes during PHYSICS TESTS~~.

In accordance with the Surveillance Frequency Control Program

LA01

Add proposed SR 3.1.8.4 ~~with a Frequency of 24 hours~~

M01

A03 **INSERT 1**

and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 16.e, may be reduced to 3 required channels,

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ITS A01 ITS 3.1.8

3.1 REACTIVITY CONTROL SYSTEMS
~~SPECIAL TEST EXCEPTIONS~~ A02

3/4.10.3 PHYSICS TESTS Exceptions – MODE 2

LIMITING CONDITION FOR OPERATION

INSERT 1 A03

LCO 3.1.8 3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

~~b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and~~ A04

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

Add proposed LCO 3.1.8.b M01

Applicability APPLICABILITY: MODE 2.

During PHYSICS TESTS initiated in A05

ACTION:

Add proposed ACTION A M01

ACTION B a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

ACTION C b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore (T_{avg}) to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

ACTION D

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER ~~at least once per hour during PHYSICS TESTS~~.

In accordance with the Surveillance Frequency Control Program LA01

SR 3.1.8.1 4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL ~~FUNCTIONAL~~ TEST prior to initiating PHYSICS TESTS.

OPERATIONAL M02

SR 3.1.8.2 4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F ~~at least once per 30 minutes during PHYSICS TESTS~~.

In accordance with the Surveillance Frequency Control Program LA01

Add proposed SR 3.1.8.4 ~~with a Frequency of 24 hours~~ M01

A03 **INSERT 1**

and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 16.e, may be reduced to 3 required channels,

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DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS Section 3.10 is titled SPECIAL TEST EXCEPTIONS. CTS Specification 3.10.3 is titled PHYSICS TESTS. ITS Section 3.1 is titled REACTIVITY CONTROL SYSTEMS. ITS Specification 3.1.8 is titled PHYSICS TESTS Exceptions – MODE 2. This changes the CTS by changing the title of the Section and the Specification.

This change is acceptable because the requirements have not changed. This change is to the titles only. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.10.3 states the limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS. ITS LCO 3.1.8 includes an allowance to reduce the required number of channels for ITS LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," Function 2 (Power Range Neutron Flux), Function 3 (Power Range Neutron Flux Rate), Function 6, (Overtemperature ΔT), and Function 16.e (Power Range Neutron Flux, P-10) from "4" to "3." This changes CTS 3.10.3 by adding an allowance to reduce the number of required RTS channels from "4" to "3" for specified Functions.

The purpose of CTS 3.10.3 is to allow some flexibility during the performance of PHYSICS TESTS while ensuring appropriate limitations are in place to help ensure safe operation. This change is acceptable because the minimum channels required for OPERABILITY for these RTS Functions in CTS Table 3.3-1 is currently "3." This allowance is needed since the "Required Channels" in ITS 3.3.1, Reactor Trip System Instrumentation, is "4." The change from CTS "MINIMUM CHANNELS OPERABLE" to ITS "Required Channels is discussed in Discussion of Changes for ITS 3.3.1. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 3.10.3.b states that the limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS provided the reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER. ITS 3.1.8 states the requirements of certain Specifications may be suspended but contains no requirements on the Intermediate and Power Range Channels. The ITS contains the same requirements on the Intermediate and Power Range Channels in ITS LCO 3.3.1. This changes the CTS by eliminating the requirement that the Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range

DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

Channels are set at ≤ 25% of RATED THERMAL POWER from the test exception.

This change is acceptable because the Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are contained in ITS LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Repeating that requirement in the test exception LCO is unnecessary. This change is designated as administrative as it eliminates a repeated requirement from the CTS, resulting in no technical change to the CTS.

A05 CTS 3.10.3 is applicable in MODE 2. ITS 3.1.8 is applicable during PHYSICS TESTS initiated in MODE 2. This changes the CTS such that the Specification is applicable in MODE 2 only when a PHYSICS TEST is initiated.

The purpose of ITS 3.1.8 Applicability is to ensure the ACTIONS contained in the Specification are followed. The wording of the CTS appears to be contradictory because, if THERMAL POWER exceeds 5% RTP, then the test exception Specification Applicability is exited and the Actions no longer apply. However, it is clear that the CTS Action should be applied if THERMAL POWER exceeds 5% RTP and PHYSICS TESTS are in progress. The ITS Applicability eliminates this apparent contradiction and allows the test exception Conditions and Required Actions to be applied when the LCO is not met. This is consistent with the wording of the CTS ACTION. This change is designated as administrative because it clarifies the current wording of the Specification with no change in intent.

MORE RESTRICTIVE CHANGES

M01 CTS 3.10.3 states that limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS and provides restrictions that must be followed when utilizing the CTS exception. ITS 3.1.8 adds a requirement that SHUTDOWN MARGIN must be within the limits provided in the COLR. A Surveillance (ITS SR 3.1.8.4), to verify the SHUTDOWN MARGIN every 24 hours, and an ACTION (ITS 3.1.8 ACTION A), to follow if the SHUTDOWN MARGIN is not met, are also added. See DOC LA01 for the discussion on moving the 24 hours Frequency to the Surveillance Frequency Control Program. This changes the CTS by imposing an additional requirement on the application of the test exception LCO.

This change is acceptable because it imposes reasonable restrictions on the performance of PHYSICS TESTS when the control rod and RCS minimum temperature Specifications are allowed to be violated. The Bases for ITS 3.1.1, "SHUTDOWN MARGIN," states that during MODE 2, the SHUTDOWN MARGIN is ensured by compliance with the rod insertion limit Specifications. Under this test exception, those limits are allowed to be violated. This change is designated as more restrictive because it imposes additional restrictions not found in the CTS.

M02 CTS 4.10.3.2 requires performance of a CHANNEL FUNCTIONAL TEST on each Intermediate and Power Range Channel. ITS SR 3.1.8.1 requires

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performance of a CHANNEL OPERATIONAL TEST (COT) on each intermediate and power range channel. This changes the CTS by requiring a COT instead of a CHANNEL FUNCTIONAL TEST.

CTS defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated signal into the sensor as close to the sensor as practicable to verify OPERABILITY. ITS defines a COT as the injection of an actual or simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. This changes the CTS by requiring adjustments of the setpoints so that the Intermediate and Power Range Channel are within the necessary range and accuracy. This change is designated as more restrictive because it imposes additional requirements on testing.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.10.3.1 requires determining that the THERMAL POWER is less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS. CTS 4.10.3.3 requires determining that the Reactor Coolant System temperature (T_{avg}) is greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS. ITS SR 3.1.8.2 and ITS SR 3.1.8.3 requires similar Surveillances and specifies the periodic Frequencies as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated

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DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

3.10.3

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient~~,~~";
LCO 3.1.4, "Rod Group Alignment Limits~~,~~";
LCO 3.1.5, "Shutdown Bank Insertion Limits~~,~~";
LCO 3.1.6, "Control Bank Insertion Limits~~,~~"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

4

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 1~~8~~6.e, may be reduced to 3 required channels, provided:

2

a. RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F,

1

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

Applicability

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01	A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
ACTION a	B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
ACTION b	C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.10.3.2	SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per ~~[~~SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1~~]~~.	Prior to initiation of PHYSICS TESTS	1
4.10.3.3	SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 3 3
4.10.3.1	SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M01	SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

3.10.3 LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient~~,~~";
LCO 3.1.4, "Rod Group Alignment Limits~~,~~";
LCO 3.1.5, "Shutdown Bank Insertion Limits~~,~~";
LCO 3.1.6, "Control Bank Insertion Limits~~,~~"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality" 4

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 1~~8~~6.e, may be reduced to 3 required channels, provided: 2

a. RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F, 1

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

Applicability APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01	A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
ACTION a	B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
ACTION b	C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.10.3.2	SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per ~~[~~SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1~~]~~.	Prior to initiation of PHYSICS TESTS	1
4.10.3.3	SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 3 3
4.10.3.1	SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
DOC M01 SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8, PHYSICS TEST EXCEPTIONS – MODE 2

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.1.8.2, SR 3.1.8.3, and SR 3.1.8.4 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

4. The punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. The ~~All~~ functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2). 5

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,

b. Validate the analytical models used in the design and analysis,

c. Verify the assumptions used to predict unit response,

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include the ~~all~~ information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation. 5

BASES

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn~~,~~ ; 6

b. Critical Boron Concentration - Control Rods Inserted~~;~~ ;

c. Control Rod Worth~~;~~ ; and 1

d. Isothermal Temperature Coefficient (ITC)~~, and~~ . 1

~~e. Neutron Flux Symmetry.~~ 1

~~The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2~~. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. 1

[a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs. 2

b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% Δk/k when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

BASES

BACKGROUND (continued)

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

~~e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry~~ 1

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BASES

BACKGROUND (continued)

~~Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.]~~ 1 2

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Core Operating Limit Methodology for Westinghouse Designed PWRs ~~Westinghouse Reload Safety Evaluation Methodology Report~~ (Ref. 5). 1 The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The U FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table~~s~~ ~~[14.1-1 and~~ 14.1-2~~]~~ summarize s the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-~~1985~~ 1997 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for ~~all~~ the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 531°F, and SDM is within the limits provided in the COLR. 1 2 5 1 5

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, ~~which represent~~ representing initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), ~~which~~ that are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. 5

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

~~Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.~~ 1

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One power range neutron flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 1~~8~~6 may be reduced to 3 required channels during the performance of PHYSICS TESTS provided: 1

a. RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F, 2

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. ~~[~~ Verification of the RCS temperature ~~at a Frequency of 30 minutes~~ during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated. 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 4 3

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. ~~[~~ Verification of the THERMAL POWER ~~at a Frequency of 30 minutes~~ during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated. 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~] 4 3

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration,

b. Control bank position,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration,

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH),

h. ~~Moderate~~ Moderator temperature defect, when above the POAH, and 1

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

[~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4 3

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-~~1985, December 13, 1985~~ 1997. 1

5. ~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985~~ BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse Designed PWRs," June 1989. 1

~~6. WCAP-11618, including Addendum 1, April 1989.~~ 1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. The ~~All~~ functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2). 5

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,

b. Validate the analytical models used in the design and analysis,

c. Verify the assumptions used to predict unit response,

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include the ~~all~~ information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation. 5

BASES

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn~~,~~ ; 6

b. Critical Boron Concentration - Control Rods Inserted~~;~~ ;

c. Control Rod Worth~~;~~ ; and 1

d. Isothermal Temperature Coefficient (ITC)~~, and~~ . 1

~~e. Neutron Flux Symmetry.~~ 1

~~The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2~~. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. 1

[a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs. 2

b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

BASES

BACKGROUND (continued)

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

~~e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry~~ 1

BASES

BACKGROUND (continued)

~~Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.]~~ 1 2

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the ~~Westinghouse Reload Safety Evaluation Methodology Report~~ Core Operating Limit Methodology for Westinghouse Designed PWRs (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations. 1

The UFSAR defines requirements 3 for initial testing of the facility, including PHYSICS TESTS. Table~~s~~ ~~[~~14.1-~~1 and 14.1-2]~~ summarizes the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-~~1985~~ 1997 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for ~~all~~ the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 531°F, and SDM is within the limits provided in the COLR. 1 2 5 1 5

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, ~~which represent~~ representing initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), ~~which~~ that are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. 5

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

~~Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.~~ 1

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One power range neutron flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 1~~8~~6 may be reduced to 3 required channels during the performance of PHYSICS TESTS provided: 1

a. RCS lowest loop average temperature is ≥ ~~[~~531~~]~~°F, 2

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. ~~[~~ Verification of the RCS temperature ~~at a Frequency of 30 minutes~~ during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated. 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~-----]~~ 3

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. ~~[~~ Verification of the THERMAL POWER ~~at a Frequency of 30 minutes~~ during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated. 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~] 4 3

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration,

b. Control bank position,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration,

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH),

h. Moderator temperature ~~Moderate~~ defect, when above the POAH, and 1

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

[~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.~~

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~------------------------------REVIEWER'S NOTE------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4 3

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August, 1978.
4. ANSI/ANS-19.6.1-1997 ~~1985, December 13, 1985~~. 1
5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse Designed PWRs," June 1989 ~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985~~. 1

~~6. WCAP-11618, including Addendum 1, April 1989.~~ 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8 BASES, PHYSICS TESTS EXCEPTIONS – MODE 2

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS SR 3.1.8.2, SR 3.1.8.3, and SR 3.1.8.4 Bases provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Editorial changes made for enhanced clarity/consistency.

6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 9

Relocated/Deleted Current Technical Specifications (CTS)

CTS 3/4.10.1, SHUTDOWN MARGIN

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

CTS 3/4.10.1

~~3/4.10 SPECIAL TEST EXCEPTIONS~~

~~3/4.10.1 SHUTDOWN MARGIN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).~~

~~APPLICABILITY: MODE 2.~~

~~ACTION:~~

~~a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~

~~b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~

M01

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.~~

~~4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.~~

CTS 3/4.10.1

~~3/4.10 SPECIAL TEST EXCEPTIONS~~

~~3/4.10.1 SHUTDOWN MARGIN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).~~

~~APPLICABILITY: MODE 2.~~

~~ACTION:~~

~~a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~

M01

~~b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 35 gpm of a solution containing greater than or equal to 6120 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.~~

~~4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.~~

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DISCUSSION OF CHANGES
CTS 3/4.10.1, SHUTDOWN MARGIN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M01 CTS 3.10.1 provides an exception to the SHUTDOWN MARGIN requirements in CTS 3.1.1.1 in MODE 2 due to the purpose of the measurement of rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s). According to the Bases, this special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. This changes the CTS by eliminating a special test exception.

This change is acceptable because this method of testing is no longer used. As a result, the CTS special test exception is not needed. Other rod worth measurement techniques that do not violate the SHUTDOWN MARGIN requirements are used. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.1, SHUTDOWN MARGIN

There are no specific No Significant Hazards Considerations for this Specification.

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CTS 3/4.10.2, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

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

CTS 3/4.10.2

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:~~

~~a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and~~

~~b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.~~

~~APPLICABILITY: MODE 1~~

~~ACTION:~~

~~With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specification 3.13.1., 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:~~

M01

~~a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or~~

~~b. Be in HOT STANDBY within 6 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.~~

~~4.10.2.2 Perform the surveillance required by the below listed Specifications at least once per 12 hours during PHYSICS TESTS:~~

~~a. Specification 4.2.2.2 and 4.2.2.3~~

~~b. Specification 4.2.3.2.~~

CTS 3/4.10.2

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:~~

~~a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and~~

~~b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.~~

~~APPLICABILITY: MODE 1.~~

~~ACTION:~~

~~With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:~~

M01

~~a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or~~

~~b. Be in HOT STANDBY within 6 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.~~

~~4.10.2.2 Perform the surveillance required by the below listed Specifications at least once per 12 hours during PHYSICS TESTS:~~

~~a. Specification 4.2.2 2 and 4.2.2.3~~

~~b. Specification 4.2.3.2~~

DISCUSSION OF CHANGES
CTS 3/4.10.2, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M01 CTS 3/4.10.2 provides an exception to the rod group height, rod insertion, and power distribution limits specifications. This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions. The ITS does not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because these types of PHYSICS TESTS (measurement of control rod worth and determination of the reactor stability index as well as the damping factor under xenon oscillation conditions) are only performed during initial plant startup test programs. These tests are not performed during post-refueling PHYSICS TESTS. As a result, the CTS special test exception is not needed. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.2, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

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CTS 3/4.10.4, REACTOR COOLANT LOOPS

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

CTS 3/4.10.1

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.4 REACTOR COOLANT LOOPS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS provided:~~

~~a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and~~

~~b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER~~

~~APPLICABILITY: During operation below the P-7 Interlock Setpoint.~~

M01

~~ACTION:~~

~~With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.~~

~~4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating startup or PHYSICS TESTS.~~

CTS 3/4.10.1

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN~~

A01

~~3.10.5 This specification is deleted.~~

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CTS 3/4.10.4

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.4 REACTOR COOLANT LOOPS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:~~

~~a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and~~

~~b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER.~~

~~APPLICABILITY: During operation below the P-7 Interlock Setpoint.~~

M01

~~ACTION:~~

~~With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during start up and PHYSICS TESTS.~~

~~4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating start up and PHYSICS TESTS.~~

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CTS 3/4.10.4

~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN~~

A01

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.5 This specification is deleted.~~

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DISCUSSION OF CHANGES
CTS 3/4.10.4, REACTOR COOLANT LOOPS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 3/4.10.4 provides an exception to the reactor coolant loops Specification. This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels. Testing within the required frequency is sufficient for verification that the power range and intermediate range monitors are properly functioning. The ITS does not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because these types of PHYSICS TESTS are no longer performed. Future PHYSICS TESTS will be performed under 3.1.8, "PHYSICS TESTS Exceptions – MODE 2." As a result this CTS Special test exception is not needed. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.4, REACTOR COOLANT LOOPS

There are no specific No Significant Hazards Considerations for this Specification.

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ENCLOSURE 2

VOLUME 7

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.2
POWER DISTRIBUTION LIMITS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR (F_Q(X,Y,Z))

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

A01

ITS

ITS 3.2.1

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(X,Y,Z)$

LIMITING CONDITION FOR OPERATION

LCO 3.2.1 3.2.2 $F_Q(X,Y,Z)$ shall be maintained within the acceptable limits specified in the COLR:

Applicability APPLICABILITY: MODE 1

ACTION:

Add proposed ACTION A Note M01

ACTION A With ~~$F_Q(X,Y,Z)$~~ exceeding its limit: steady state

$F_Q^C(X,Y,Z)$ A02

Required Action A.1 a. Reduce THERMAL POWER at least 1% for each 1% ~~$F_Q(X,Y,Z)$~~ exceeds the limit within 15 minutes, and similarly reduce the following:

after each $F_Q(X,Y,Z)$ determination M02

Required Action A.2 1. Administratively reduce the allowable power at each point along the AFD limit lines within 2 hours, and

72 L01

Required Action A.4 2. The Power Range Neutron Flux-High Trip Setpoints within the next ~~4~~ hours.

after each $F_Q(X,Y,Z)$ determination M02

Required Action A.3 b. POWER OPERATION may proceed for up to 48 hours. Subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints ~~(value of K_4)~~ have been reduced at least 1% ~~(in ΔT span)~~ for each 1% that ~~$F_Q(X,Y,Z)$~~ exceeds the limit specified in the COLR.

LA01

$F_Q^C(X,Y,Z)$ A02

Required Action A.5 c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by Action a. and b., above; THERMAL POWER may then be increased provided ~~$F_Q(X,Y,Z)$~~ is demonstrated ~~through incore mapping~~ to be within its limits.

LA02

Add proposed ACTION D M03

SURVEILLANCE REQUIREMENTS

SR NOTE 4.2.2.1 The provisions of Specification 4.0.4 are not applicable. M04

A01

ITS

ITS 3.2.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.1.2
SR 3.2.1.3

4.2.2.2 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if $F_Q(X,Y,Z)$ is within its limit by:

~~a. Using the moveable incore detectors to obtain a power distribution map ($F_Q^M(X,Y,Z)$ *) at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Satisfying the following relationship:~~

$\cancel{F_Q^M(X,Y,Z) \leq BQNOM(X,Y,Z)}$

~~where BQNOM (X,Y,Z)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.~~

~~The BQNOM (X,Y,Z) factors are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:~~

~~1. Lower core region from 0 to 15%, inclusive.~~

~~2. Upper core region from 85 to 100%, inclusive.~~

LA03

c. If the above relationship is not satisfied, then

1. For that location, calculate the % margin to the maximum allowable design as follows:

SR 3.2.1.2

$$\% \text{ AFD Margin} = \cancel{\left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) \text{x } 100\%}$$

SR 3.2.1.3

$$\% \ f_2(\Delta I) \text{ Margin} = \cancel{\left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) \text{x } 100\%}$$

LA03

~~where BQDES(X,Y,Z)** and BCDES(X,Y,Z)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within Limiting Condition for Operation limits, and include allowances for the calculational and measurement uncertainties.~~

LA03

~~* No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$, because the limits include uncertainties.~~

~~** BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.~~

LA03

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A01

ITS ITS 3.2.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

~~2. Find the minimum margin of all locations examined in 4.2.2.2.c.1 above.~~

~~AFD min margin = minimum % margin value of all locations examined.~~

~~$f_2(\Delta I)$ OPΔT min margin = minimum % margin value of all locations examined.~~ LA03

ACTION B 3. If the AFD min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, ~~or the action statements for 3.2.2 shall be followed.~~ M05

REQUIRED ACTION B.2 (a) Within 2 hours, administratively reduce the negative AFD limit lines at each power level by:

~~Reduced AFD^{Limit} = (AFD^{Limit} from COLR) + absolute value of (NSLOPE^{AFD}* % x AFD min margin of 4.2.2.2.c.2)~~ LA03

REQUIRED ACTION B.1 (b) Within 2 hours, administratively reduce the positive AFD limit lines at each power level by:

~~Reduced AFD^{Limit} = (AFD^{Limit} from COLR) absolute value of (PSLOPE^{AFD}* % X AFD min margin)~~ LA03

ACTION C 4. If the $f_2(\Delta I)$ min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, ~~or the action statements for 3.2.2 shall be followed.~~ M05

REQUIRED ACTION C.2 (a) Within 48 hours, reduce the OPΔT negative $f_2(\Delta I)$ breakpoint limit by:

~~Reduced OPΔT negative $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) + absolute value of~~

~~(NSLOPE$^{f_2(\Delta I)**}$ % x $f_2(\Delta I)$ min margin)~~ LA03

REQUIRED ACTION B.1/B.2 * NSLOPE^{AFD} and PSLOPE^{AFD} are the amount of AFD adjustment required to compensate for each 1% that F_Q(X,Y,Z) exceeds the limit ~~provided in the COLR per Specification 6.9.1.14~~ LA03

REQUIRED ACTION C.1/C.2 ** NSLOPE$^{f_2(\Delta I)}$ and PSLOPE$^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that F_Q(X,Y,Z) exceeds the limit ~~provided in the COLR per Specification 6.9.1.14~~ LA03

A01

ITS | ITS 3.2.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

REQUIRED ACTION C.1

(b) Within 48 hours, reduce the OPΔT positive $f_2(\Delta I)$ breakpoint limit by:

~~Reduced OPΔT positive $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) absolute value of~~ $(\mathrm{PSLOPE}^{f_2(\Delta I)}{**}\%\ \mathrm{x}\ f_2(\Delta I)\ \mathrm{min\ margin})$ LA03

d. Measuring $F_Q^M(X,Y,Z)$ according to the following schedule: INSERT 1 M06

Once within 12 hours after M07

SR 3.2.1.1
SR 3.2.1.2
SR 3.2.1.3

1. ~~Upon~~ achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined,*** ~~or~~ and M08

In accordance with the Surveillance Frequency Control Program LA04

2. ~~At least once per 31 Effective Full Power Days, whichever occurs first.~~ ~~thereafter~~ M08

SR 3.2.1.2/SR 3.2.1.3 NOTE

e. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding $F_Q^M(X,Y,Z) \succ \mathrm{BQNOM}(X,Y,Z)$, either of the following actions specified shall be taken.

SR 3.2.1.2 / SR 3.2.1.3 NOTE a.

1. $F_Q^M(X,Y,Z)$ shall be increased over that specified in 4.2.2.2.a by the appropriate factor specified in the COLR, and 4.2.2.2.c repeated, or

SR 3.2.1.2 / SR 3.2.1.3 NOTE b.

2. $F_Q^M(X,Y,Z)$ shall be evaluated according to 4.2.2.2 at or before the time when the margin is projected to result in one of the actions specified in 4.2.2.2.c.3 or 4.2.2.2.c.4.

INSERT 2 M09

SR 3.2.1.1

4.2.2.3 When ~~$F_Q(X,Y,Z)$~~ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured ~~$F_Q(X,Y,Z)$~~ shall be obtained from a power distribution map~~, increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR according to Specification 3.2.2.~~ LA02

$F_Q^C(X,Y,Z)$ A02

REQUIRED ACTION C.1/C.2

** $\mathrm{NSLOPE}^{f_2(\Delta I)}$ and $\mathrm{PSLOPE}^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit ~~provided in the COLR per Specification 6.9.1.14~~ LA03

SR Note

*** Not required to be performed until 12 hours after an equilibrium ~~During power escalation at the beginning of each cycle, power level may be increased until a~~ power level ~~for extended operation~~ has been achieved and power distribution map can be obtained. M04

CTS 3.2.1

DOC M06

INSERT 1

Once after each refueling prior to THERMAL POWER exceeding 75% RTP

DOC M09

INSERT 2

Once after each refueling prior to THERMAL POWER exceeding 75% RTP

AND

Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(X, Y, Z)$ was last verified

AND

In accordance with the Surveillance Frequency Control Program

LA04

~~At least once per 31 Effective Full Power Days~~

~~This page intentionally deleted.~~

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A01

ITS — ITS 3.2.1

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR- $F_Q(X,Y,Z)$

LIMITING CONDITION FOR OPERATION

LCO 3.2.1 — 3.2.2 $F_Q(X,Y,Z)$ shall be maintained within the acceptable limits specified in the COLR.

Applicability — APPLICABILITY: MODE 1

ACTION: — Add proposed ACTION A Note — M01

ACTION A — With ~~$F_Q(X,Y,Z)$~~ $F_Q^C(X,Y,Z)$ exceeding its steady state limit: — A02

Required Action A.1 — a. Reduce THERMAL POWER at least 1% for each 1% ~~$F_Q(X,Y,Z)$~~ $F_Q^C(X,Y,Z)$ exceeds the limit within 15 minutes after each $F_Q(X,Y,Z)$ determination, and similarly reduce the following: — M02

Required Action A.2 — 1. Administratively reduce the allowable power at each point along the AFD limit lines within 2 hours, and

Required Action A.4 — 2. The Power Range Neutron Flux-High Trip Setpoints within the next ~~4~~ 72 hours after each $F_Q(X,Y,Z)$ determination. — L01 — M02

Required Action A.3 — b. POWER OPERATION may proceed for up to 48 hours. Subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints ~~(value of K_4)~~ have been reduced at least 1% ~~(in ΔT span)~~ for each 1% that ~~$F_Q(X,Y,Z)$~~ $F_Q^C(X,Y,Z)$ exceeds the limit specified in the COLR. — LA01 — A02

Required Action A.5 — c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by Action a. and b., above; THERMAL POWER may then be increased provided ~~$F_Q(X,Y,Z)$~~ is demonstrated ~~through incore mapping~~ to be within its limits. — LA02

Add proposed ACTION D — M03

SURVEILLANCE REQUIREMENTS

SR NOTE — 4.2.2.1 The provisions of Specification 4.0.4 are not applicable. — M04

A01

ITS ITS 3.2.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.1.2
SR 3.2.1.3

4.2.2.2 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if $F_Q(X,Y,Z)$ is within its limit by:

~~a. Using the moveable incore detectors to obtain a power distribution map ($F_Q^M(X,Y,Z)$ *) at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Satisfying the following relationship:~~

$$\cancel{F_Q^M(X,Y,Z) \leq BQNOM(X,Y,Z)}$$

~~where BQNOM (X,Y,Z)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.~~

~~The BQNOM (X,Y,Z) factors are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:~~

~~1. Lower core region from 0 to 15%, inclusive.~~

~~2. Upper core region from 85 to 100%, inclusive.~~

LA03

c. If the above relationship is not satisfied, then

1. For that location, calculate the % margin to the maximum allowable design as follows:

SR 3.2.1.2

$$\% \text{ AFD Margin} = \cancel{\left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) \times 100\%}$$

LA03

SR 3.2.1.3

$$\% \ f_2(\Delta I) \text{ Margin} = \cancel{\left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) \times 100\%}$$

~~where BQDES (X,Y,Z)** and BCDES(X,Y,Z)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within Limiting Condition for Operation limits, and include allowances for the calculational and measurement uncertainties.~~

~~* No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$, because the limits include uncertainties.~~

~~** BQNOM (X,Y,Z), BQDES (X,Y,Z), and BCDES (X,Y,Z) Data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.~~

LA03

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A01

ITS

ITS 3.2.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

~~2. Find the minimum margin of all locations examined in 4.2.2.2.c.1 above.~~

~~AFD min margin = minimum % margin value of all locations examined.~~

~~$f_2(\Delta I)$ OPΔT min margin = minimum % margin value of all locations examined.~~

LA03

ACTION B 3. If the AFD min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, ~~or the action statements for 3.2.2 shall be followed.~~

M05

REQUIRED ACTION B.2 (a) Within 2 hours, administratively reduce the negative AFD limit lines at each power level by:

~~Reduced AFD^{Limit} = (AFD^{Limit} from COLR) + absolute value of ($NSLOPE^{AFD}$* % x AFD min margin of 4.2.2.2.c.2)~~

LA03

REQUIRED ACTION B.1 (b) Within 2 hours, administratively reduce the positive AFD limit lines at each power level by:

~~Reduced AFD^{Limit} = (AFD^{Limit} from COLR)-absolute value of ($PSLOPE^{AFD}$* % X AFD min margin)~~

LA03

ACTION C 4. If the $f_2(\Delta I)$ min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, ~~or the action statements for 3.2.2 shall be followed.~~

M05

REQUIRED ACTION C.2 (a) Within 48 hours, reduce the OPΔT negative $f_2(\Delta I)$ breakpoint limit by:

~~Reduced OPΔT negative $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) + absolute value of~~

~~($NSLOPE^{f_2(\Delta I)}$** % x $f_2(\Delta I)$ min margin)~~

LA03

REQUIRED ACTION B.1/B.2 * $NSLOPE^{AFD}$ and $PSLOPE^{AFD}$ are the amount of AFD adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit ~~provided in the COLR per Specification 6.9.1.14.~~

LA03

REQUIRED ACTION C.1/C.2 ** $NSLOPE^{f_2(\Delta I)}$ and $PSLOPE^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit ~~provided in the COLR per Specification 6.9.1.14.~~

LA03

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A01

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

REQUIRED ACTION C.1

(b) Within 48 hours, reduce the OPΔT positive $f_2(\Delta I)$ breakpoint limit by:

~~Reduced OPΔT positive $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) absolute value of (PSLOPE$^{f_2(\Delta I)}$ ** % x $f_2(\Delta I)$ min margin)~~ LA03

d. Measuring $F_Q^M(X,Y,Z)$ according to the following schedule: INSERT 3 M06

Once within 12 hours after M07

SR 3.2.1.1 SR 3.2.1.2 SR 3.2.1.3

1. ~~Upon~~ achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined,*** ~~or~~ and M08

In accordance with the Surveillance Frequency Control Program LA04

~~2. At least once per 31 Effective Full Power Days, whichever occurs first.~~ ~~thereafter~~ M08

SR 3.2.1.2/SR 3.2.1.3 NOTE

e. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding $F_Q^M(X,Y,Z) \succ BQNOM(X,Y,Z)$, either of the following actions specified shall be taken.

SR 3.2.1.2 / SR 3.2.1.3 NOTE a.

1. $F_Q^M(X,Y,Z)$ shall be increased over that specified in 4.2.2.2.a by the appropriate factor specified in the COLR, and 4.2.2.2.c repeated, or

SR 3.2.1.2 / SR 3.2.1.3 NOTE b.

2. $F_Q^M(X,Y,Z)$shall be evaluated according to 4.2.2.2 at or before the time when the margin is projected to result in one of the actions specified in 4.2.2.2.c.3 or 4.2.2.2.c.4.

INSERT 4 M09

SR 3.2.1.1

4.2.2.3 When ~~$F_Q(X,Y,Z)$~~ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured ~~$F_Q(X,Y,Z)$~~ shall be obtained from a power distribution map, ~~increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR according to Specification 3.2.2.~~ LA02

$F_Q^C(X,Y,Z)$ A02

REQUIRED ACTION C.1/C.2

** $NSLOPE^{f2(\Delta I)}$ and $PSLOPE^{f2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit ~~provided in the COLR per Specification 6.9.1.14.~~ LA03

SR Note

Not required to be performed until 12 hours after an equilibrium

*** ~~During power escalation at the beginning of each cycle, power level may be increased until a~~ power level ~~for extended operation~~ has been achieved and power distribution map can be obtained. M04

CTS 3.2.1

DOC M06

INSERT 3

Once after each refueling prior to THERMAL POWER exceeding 75% RTP

DOC M09

INSERT 4

Once after each refueling prior to THERMAL POWER exceeding 75% RTP

AND

Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(X,Y,Z)$ was last verified

AND

In accordance with the Surveillance Frequency Control Program

LA04

~~At least once per 31 Effective Full Power Days~~

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DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 4.2.2.2 evaluates $F_Q^M(X,Y,Z)$ to determine if $F_Q(X,Y,Z)$ is within the limits. CTS 4.2.2.3 evaluates $F_Q(X,Y,Z)$ for reasons other than meeting the requirements of CTS 4.2.2.2 and requires the overall measured $F_Q(X,Y,Z)$ be obtained from a distribution flux map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR. ITS 3.2.1 ACTION A and SR 3.2.1.1 use $F_Q^C(X,Y,Z)$ to represent the overall measured $F_Q(X,Y,Z)$ adjusted for measurement uncertainty and manufacturing tolerances. This changes the CTS by adding a new term, $F_Q^C(X,Y,Z)$ which reflects the requirements in CTS 4.2.2.3 for evaluating the steady state limit of $F_Q(X,Y,Z)$ specified in the COLR.

BAW-10163PA, "Core Operating Limit Methodology for Westinghouse-Designed PWRs" June 1989, requires that $F_Q(X,Y,Z)$ is compared against three limits: (1) steady state limit, $(F_Q^{RTP} / P) * K(Z)$, (2) limiting condition LOCA limit, BQDES(X,Y,Z), and (3) Limiting condition centerline fuel melt limit, BCDES(X,Y,Z). BAW-10163PA further states that the overall measured $F_Q(X,Y,Z)$ must be adjusted for uncertainty prior to comparison to the steady state limit.
The CTS 3.2.2 Surveillance Requirements address both the steady state and the limiting conditions. CTS 4.2.2.2, in part evaluates $F_Q^M(X,Y,Z)$ for both BQDES(X,Y,Z) and BCDES(X,Y,Z) to ensure the $F_Q(X,Y,Z)$ limit is met at limiting conditions. Thus if BQDES(X,Y,Z) and BCDES(X,Y,Z) are met, the steady state limit is met These verifications are reflected in ITS SR 3.2.1.2 and SR 3.2.1.3. CTS 4.2.2.3 addresses evaluation of the steady state limit directly using the overall measured $F_Q(X,Y,Z)$ adjusted by the two penalty factors, $F_Q^C(X,Y,Z)$. ITS 3.2.1 uses $F_Q^C(X,Y,Z)$ throughout the Specification to refer to the steady state limit. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.2.2 ACTION c states that with $F_Q(X,Y,Z)$ exceeding its limit "Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL

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DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR (F_Q(X,Y,Z))

POWER above the reduced limit required by Action a. and b., above; THERMAL POWER may then be increased provided F_Q(X,Y,Z) is demonstrated through incore flux mapping to be within its limits." However, under CTS 3.0.2, the F_Q(X,Y,Z) measurement does not have to be completed, if compliance with the LCO is restored. ITS 3.2.1 ACTION A contains a Note which states, "Required Action A.5 must be completed whenever this Condition is entered." ITS Required Action A.5 requires performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 prior to increasing THERMAL POWER above the limit of Required Action A.1. This changes the CTS by requiring $F_Q^C(X,Y,Z)$ verification to be made even if $F_Q^C(X,Y,Z)$ is restored to within its limit.

The purpose of CTS 3.2.2 ACTION c is to ensure that when F_Q(X,Y,Z) has exceeded the limit, compensatory measures are commenced to restore core power distribution to within the limits prior to increasing THERMAL POWER. This change is acceptable, because it establishes appropriate compensatory measurements for violation of the F_Q(X,Y,Z) limit. As power is reduced under ITS Required Action A.1, the margin to the F_Q(X,Y,Z) limit increases. Therefore, compliance with the LCO could be restored during the power reduction. Verifying that the limit is met as power is increased ensures that the limit continues to be met and does not remain unmeasured for up to 31 EFPD. This change is designated as a more restrictive change, because it imposes requirements in addition to those in the CTS.

M02 CTS 3.2.2 ACTION states in part that when F_Q(X,Y,Z) has exceeded the limit, to (1) Reduce THERMAL POWER at least 1% for each 1% F_Q(X,Y,Z) exceeds the limit within 15 minutes, (2) Administratively reduce the allowable power at each point along the AFD limit lines within 2 hours, (3) Reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, (4) Reduce the Overpower Delta T Trip Setpoints (value of K_4) at least 1% (in ΔT span) for each 1% that F_Q(X,Y,Z) exceeds the limit specified in the COLR within the next 48 hours, (5) Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by Action a. and b., above; THERMAL POWER may then be increased provided F_Q(X,Y,Z) is demonstrated through incore mapping to be within its limits. ITS 3.2.1 has similar Required Actions and Completion Times with the added requirement to ensure the times are met after each $F_Q^C(X,Y,Z)$ determination. This changes the CTS by requiring the Required Actions to be re-performed within a specific Completion Time after each flux map determination.

The purpose of the CTS 3.2.2 ACTIONs is to ensure that when F_Q(X,Y,Z) has exceeded the limit, compensatory measures are commenced to restore core power distribution to within the limits assumed in the safety analysis. This change is acceptable because it ensures that the Required Actions for $F_Q^C(X,Y,Z)$ not within limits will be re-performed after each $F_Q^C(X,Y,Z)$ determination within the prescribed Completion Time. When $F_Q^C(X,Y,Z)$ is not met, the margin to the limit prescribes the amount of power reduction and setpoint reduction to be performed. Therefore, each time flux mapping is performed, the determination of margin to the limit will determine if additional power reduction or additional

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ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

setpoint reduction is required. This change is designated as more restrictive, because it applies new Completion Time requirements which do not exist in the CTS.

M03 CTS 3.2.2 does not contain an Action to follow, if the provided Actions cannot be met. Therefore, CTS 3.0.3 would be entered, which would allow 1 hour to initiate a shutdown and to be in HOT STANDBY within 7 hours. ITS 3.2.1 ACTION D, states that the plant must be in MODE 2 within 6 hours, if any Required Action and associated Completion Time is not met. This changes the CTS by eliminating the one hour to initiate a shutdown and, consequently, allowing one hour less for the unit to be in MODE 2.

This change is acceptable because it provides an appropriate compensatory measure for the described conditions. If any Required Action and associated Completion Time cannot be met, the unit must be placed in a MODE in which the LCO does not apply. The LCO is applicable in MODE 1. Requiring a shutdown to MODE 2 is appropriate in this condition. The one hour allowed by CTS 3.0.3 to prepare for a shutdown is not needed, because the operators have had time to prepare for the shutdown while attempting to follow the Required Actions and associated Completion Times. This change is designated as more restrictive because it allows less time to shut down than does the CTS.

M04 CTS 4.2.2.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. CTS 4.2.2.2.d.1 Note *** states that during power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map obtained. ITS 3.2.1 has a similar note for the beginning of each cycle, however, there is no specific allowance for changing MODES at any other time with ITS LCO 3.2.1 not met. ITS LCO 3.0.4 requires, in part, that when an LCO is not met, entry into a MODE or other specified condition in the applicability shall only be made: If part a. or part b. or part c. is met. Part c allows, when an allowance is stated in the individual value, parameter or other specification. ITS 3.2.1 Surveillance Requirements Note will provide an allowance whereby, Surveillance performance is not required until 12 hours after an equilibrium power level has been achieved, at which a power distribution map can be obtained. This changes CTS by allowing entry into the MODE of Applicability by only deferring the performance of the Surveillance Requirements instead of deferring compliance with the LCO.

The purpose of CTS 4.2.2.1 is to provide an allowance for entering the next higher MODE of applicability when any Surveillance is not met. This change is acceptable because ITS provides an allowance to enter the MODE of Applicability at any time LCO 3.2.1 is not met solely based on Surveillance performance. SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level (~40%-50%). The surveillances cannot be performed in MODE 2. This change is designated as more restrictive because the CTS 4.0.4 MODE

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DISCUSSION OF CHANGES
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change allowance with the LCO not met is now limited to the performance of the SRs and does not include the allowance to change MODES for non-compliance with the acceptance criteria.

M05 CTS 3.2.2 provides two acceptable alternatives for the AFD min margin and $f_2(\Delta I)$ min margin not met. CTS 4.2.2.2.c.3 states, "If the AFD min margin in 4.2.2.2.c.2 above is < 0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed." CTS 4.2.2.2.c.4 states, "If the $f_2(\Delta I)$ min margin in 4.2.2.2.c.2 above is < 0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed." CTS 4.2.2.2.c.3.a and CTS 4.2.2.2.c.3.b have been replaced by ITS 3.2.1 Required Actions B.1 and B.2. Similarly, CTS 4.2.2.2.c.4.a and 4.2.2.2.c.4.b have been replaced with ITS 3.2.1 Required Actions C.1 and C.2. However, in both cases the option for, "the action statements for 3.2.2 shall be followed" has not been retained. This changes the CTS by removing the option to follow the action statement of CTS 3.2.2 for either min margin (AFD or $f_2(\Delta I)$) not met.

The purpose of CTS 4.2.2.2.c.3 and CTS 4.2.2.2.c.4 is to provide acceptable alternatives for the required compensatory actions when either AFD min margin or $f_2(\Delta I)$ min margin is not met. The CTS surveillance requirements for either AFD min margin or $f_2(\Delta I)$ min margin not met require either the administrative reduction in their respective setpoints or the option of entering the actions of LCO 3.2.2. The CTS Actions for 3.2.2, $F_Q(X,Y,Z)$ exceeding the limits, require in part the reduction of THERMAL POWER, reduction of AFD limit lines, and reduction $f_2(\Delta I)$ breakpoint limits. ITS 3.2.1 has removed this option, but retains the requirement for administrative reduction in AFD limits, ITS CONDITION B, or $f_2(\Delta I)$ breakpoint limits, ITS CONDITION C. If the ITS Required Actions to administratively reduce the respective setpoints is not performed within the allowed Completion Time, Condition D will be entered requiring the Unit to be placed in MODE 2. This change is designated as more restrictive because an acceptable alternative Required Action available in CTS is being removed.

M06 CTS 4.2.2.2.d requires $F_Q(X,Y,Z)$ to be determined to be within its limit upon achieving equilibrium conditions after exceeding by 10 percent or more of RTP, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined, or at least once per 31 EFPD, whichever occurs first. ITS SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3 collectively verify that $F_Q(X,Y,Z)$ is within its limit after each refueling prior to THERMAL POWER exceeding 75% RTP, once within 12 hours after achieving equilibrium conditions after exceeding, by greater than or equal to 10% RTP, the THERMAL POWER at which $F_Q^C(X,Y,Z)$ and $F_Q^M(X,Y,Z)$ was last verified, and in accordance with the Surveillance Frequency Control Program. This changes the CTS by adding a new Frequency (Once after each refueling prior to THERMAL POWER exceeding 75% RTP). The replacement of the words "whichever occurs first" with the word "thereafter" to the Frequency is discussed in DOC M08. Moving the "31 EFPD thereafter" Frequency to the Surveillance Frequency Control Program is discussed in DOC LA04. The addition of "once within 12 hours" is discussed in DOC M07.

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ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

The purpose of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 is to verify that $F_Q(X,Y,Z)$ is within the limits assumed in the safety analysis. This change is acceptable, because adopting the new Frequency of confirming $F_Q^C(X,Y,Z)$ and $F_Q^M(X,Y,Z)$ are within the limits prior to exceeding 75% RTP following each core reload, will ensure that some determination of $F_Q^C(X,Y,Z)$ and $F_Q^M(X,Y,Z)$ is made at a lower power level at which adequate margin is available, before going to 100% RTP. This change is designated as more restrictive, because it applies new requirements which do not exist in the CTS.

M07 CTS 4.2.2.2.d requires $F_Q(X,Y,Z)$ to be determined to be within its limit upon achieving equilibrium conditions after exceeding by 10 percent or more of RTP, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined, or at least once per 31 EFPD, whichever occurs first. ITS SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3 collectively verify that $F_Q(X,Y,Z)$ is within its limit after each refueling prior to THERMAL POWER exceeding 75% RTP, once within 12 hours after achieving equilibrium conditions after exceeding, by greater than or equal to 10% RTP, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last verified, and in accordance with the Surveillance Frequency Control Program. This changes the CTS by modifying the existing Frequency (Upon achieving equilibrium conditions…) by adding a specific time element (Once within 12 hours after achieving equilibrium conditions) which limits the time duration allowed for completing a single performance after a ≥10% RTP change . The replacement of the words "whichever occurs first" with the word "thereafter" to the Frequency is discussed in DOC M08. The relocation of the "31 EFPD thereafter" Frequency to the Surveillance Frequency Control Program is discussed in DOC LA04. The addition of new Frequency (Once after each refueling prior to THERMAL POWER exceeding 75% RTP) is discussed in DOC M06.

The purpose of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 is to verify that $F_Q(X,Y,Z)$ is within the limits assumed in the safety analysis. This change is acceptable, because modifying the existing frequency by adding a specific time element completing a single performance after a ≥10% RTP change is made ensures adequate margin is available, before going to a higher power level. This change is designated as more restrictive, because it applies new requirements which do not exist in the CTS.

M08 CTS 4.2.2.2.d.1 Surveillance states "required to be performed upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined, or at least once per 31 EFPD, whichever occurs first." ITS SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 are similar, but the logical connector between the Frequencies is an "<u>AND</u>" not an "or". Additionally, the ITS 31 EFPD Frequency is qualified with "thereafter". This changes the CTS by (1) removing the phrase, "whichever occurs first" and replacing it with "thereafter" and (2) changing the CTS logical connector from "or" to "<u>AND</u>".

The purpose of CTS 4.2.2.2 is to establish both when and how often $F_Q^M(X,Y,Z)$ is measured. The intent of the CTS Frequency logical connector "or" does not provide an exclusion to perform either the situational performance or the

DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

repetitive performance of the test, because both are continuously applicable when $F_Q^M(X,Y,Z)$ is measured. Additionally, the CTS Frequency describes "when" the first performance is required (i.e. whichever occurs first) based on plant conditions. This change is acceptable because the ITS use of "AND" will ensure both the situational and periodic performances are continuously applicable. This change is designated more restrictive because the Surveillance Requirements will be required to be performed more frequently than is required in CTS.

M09 CTS 4.2.2.3 states that when $F_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(X,Y,Z)$ shall be obtained from a power distribution map, increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR according to Specification 3.2.2. Proposed ITS SR 3.2.1.1, verifies $F_Q^C(X,Y,Z)$ is within the steady state limits, (1) Once after each refueling prior to THERMAL POWER exceeding 75% RTP, and (2) Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(X,Y,Z)$ was last verified, and (3) In accordance with the Surveillance Frequency Control Program. This changes the CTS from a 4.2.2.3 measurement of $F_Q(X,Y,Z)$ to be within limits on a situational Frequency basis to the ITS Frequency of (1) Once after each refueling prior to THERMAL POWER exceeding 75% RTP, and (2) Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(X,Y,Z)$ was last verified, and (3) In accordance with the Surveillance Frequency Control Program. (The relocation of "31 EFPD, thereafter to the Surveillance Frequency Control program" is discussed in DOC LA04 (Refer to DOC A02 for the discussion of the addition of a new term describing the steady state limit, $F_Q^C(X,Y,Z)$).

The purpose of CTS 4.2.2.3 is to evaluate $F_Q(X,Y,Z)$ during those situational conditions where core power distribution limits may exceed limits assumed in the safety analysis. BAW-10163PA "Core Operating Limit Methodology for Westinghouse-Designed PWRs" June 1989 requires the measured $F_Q(X,Y,Z)$ to be compared against the steady state limit (ITS SR 3.2.1.1) and the two transient limits BQDES(X,Y,Z)(ITS SR 3.2.1.2) and BCDES(X,Y,Z)(ITS SR 3.2.1.3). ITS SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 will be performed at the same Frequencies. This change is designated as more restrictive because the situational testing Frequency of CTS 4.2.2.3 is being replaced with two new situational Frequencies and a periodic performance, once every 31 EFPD.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

REMOVED DETAIL CHANGES

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.2 ACTION b requires within 48 hours when $F_Q(X,Y,Z)$ is not within limits to reduce the Overpower Delta T Trip setpoints (value of K_4) at least 1% (in ΔT span) for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR. ITS LCO 3.2.1 Required Action A.3 requires within 48 hours of discovery that $F_Q^C(X,Y,Z)$ is not within limits, to reduce Overpower ΔT trip setpoints at least 1% for each 1% that $F_Q^C(X,Y,Z)$ exceeds the limit. This changes the CTS by moving the specific information regarding the terms, "value of K_4" and "in ΔT span," to the COLR.

The removal of these details for performing actions from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements to reduce Overpower ΔT trip setpoints at least 1% for each 1% that $F_Q^C(X,Y,Z)$ exceeds the limit. Also, this change is acceptable because the removed information will be adequately controlled in the COLR requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.2 ACTION c requires $F_Q(X,Y,Z)$ to be determined to be within its limit through incore mapping. CTS 4.2.2.3 requires $F_Q(X,Y,Z)$ to be determined to be within its limit by obtaining a power distribution map and applying manufacturing tolerances and measurement uncertainty factors before comparing the results to the $F_Q(X,Y,Z)$ limit specified in the COLR. ITS 3.2.1 Required Action A.5 and ITS SR 3.2.1.1 require verification that $F_Q^C(X,Y,Z)$ is within its limit. This changes the CTS by moving the manner in which the $F_Q(X,Y,Z)$ determination is performed to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to determine $F_Q(X,Y,Z)$ is within its limit. Also, this change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR (F_Q(X,Y,Z))

LA03 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.2.2.2, 4.2.2.2.a, 4.2.2.2.b, 4.2.2.2.*, 4.2.2.2.**, 4.2.2.2.c.1, 4.2.2.2.c.2, 4.2.2.2.c.3.a, 4.2.2.2.c.3.b, 4.2.2.2.c.4.a, 4.2.2.2.c.4.b, 4.2.2.2.d, and 4.2.2.3 provide details for evaluating F^M_Q(X,Y,Z) to determine if F_Q(X,Y,Z) is within limits. ITS SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 collectively verify that F_Q(X,Y,Z) (as discussed in DOC A4) is within limits specified in the COLR. This changes the CTS by moving the details for evaluating F^M_Q(X,Y,Z) to determine if F_Q(X,Y,Z) is within limits to the ITS Bases.

The removal of these details from the Technical Specifications and their relocation into the ITS Bases is acceptable, because the procedural steps and further details for making a determination that F_Q(X,Y,Z) is within its limits is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS Surveillance Requirement to verify F_Q(X,Y,Z) is within its limits will more closely align with the LCO requirement for F_Q(X,Y,Z) to be within the limits specified in the COLR. Also, this change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.2.2.2 requires, in part, a determination that F_Q(X,Y,Z) is within its limits at least once per 31 EFPD. ITS SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 require a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR (F_Q(X,Y,Z))

LESS RESTRICTIVE CHANGES

L01 (Category 3 – Relaxation of Completion Time) CTS 3.2.2 ACTION a.2 states, in part, that when F_Q(X,Y,Z) exceeds its limit, reduce the Power Range Neutron Flux - High Trip setpoints within the next 4 hours. ITS 3.2.1 Required Actions A.4 states with $F_Q^C(X,Y,Z)$ not within limit, reduce the Power Range Neutron Flux - High Trip setpoints by ≥ 1% for each 1% $F_Q^C(X,Y,Z)$ exceeds the limit. The ITS 3.2.1 Required Action A.4 Completion Time is "within 72 hours after each $F_Q^C(X,Y,Z)$ determination." This changes the CTS by increasing the time allowed to reduce the trip setpoints.

The purpose of CTS 3.2.2 ACTION a.2 is to lower the Power Range Neutron Flux - High Trip setpoints, which ensures continued operation is at an acceptably low power level with an adequate DNBR margin and avoids violating the $F_Q^C(X,Y,Z)$ limit. This change is acceptable, because the Completion Time is consistent with safe operation and recognizes that the safety analysis assumptions are satisfied once power is reduced, and considers the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Time allows the Power Range Neutron Flux - High Trip setpoints to be reduced in a controlled manner without challenging operators, technicians, or plant systems. Following a significant power reduction, a time period of 24 hours is allowed to reestablish steady state xenon concentration and power distribution and to take and analyze a flux map. If it is determined that $F_Q^C(X,Y,Z)$ is still not within its limit, reducing the Power Range Neutron Flux - High Trip Setpoints can be accomplished within a few hours. Furthermore, setpoint changes should only be required for extended operation in this condition, because of the risk of a plant trip during the adjustment. This change is designated as less restrictive, because additional time is allowed to lower the Power Range Neutron Flux - High Trip setpoints than was allowed in the CTS.

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

CTS

F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 1 2
3.2.1~~B~~

3.2 POWER DISTRIBUTION LIMITS

3.2.1~~B~~ Heat Flux Hot Channel Factor (F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 2 1

3.2.2 LCO 3.2.1~~B~~ F_Q(~~Z~~ X,Y,Z)~~, as approximated by F^C_Q(Z) and F^W_Q(Z),~~ shall be within the limits specified in the COLR. 2 1

Applicability APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a DOC M01 DOC M02 A. ------------NOTE------------ 5 Required Action A.~~4~~ shall be completed whenever this Condition is entered. --------------------------------- ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ not within the steady state limit. DOC L01 ACTION a.2 DOC M02	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ exceeds limit. INSERT 1 AND A.~~2~~ 4 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ exceeds limit. AND A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ exceeds limit. AND A.~~4~~ 5 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3.	15 minutes after each ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ determination 3 1 3 72 hours after each ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ determination 3 1 ~~72~~ 48 hours after each ~~F^C_Q(Z)~~ $F_Q^C(X,Y,Z)$ determination 3 1 Prior to increasing THERMAL POWER above the limit of Required Action A.1 3 4

ACTION b DOC M02

ACTION c DOC M02

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CTS

3.2.1

3 **INSERT 1**

CONDITION	REQUIRED ACTION	COMPLETION TIME
	AND A.2 Reduce, by administrative means, AFD limits ≥ 1% for each 1% $F_Q^C(X, Y, Z)$ exceeds limit.	2 hours after each $F_Q^C(X, Y, Z)$ determination

ACTION a.1
DOC M02

CTS

F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~
3.2.1~~B~~

1 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. ------------NOTE------------ Required Action B.4 shall be completed whenever this Condition is entered. ---------------------------------~~ ~~$F_Q^W(Z)$ not within limits.~~	~~B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.~~ ~~AND~~ ~~B.2 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.~~ ~~AND~~ ~~B.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.~~ ~~AND~~ ~~B.4 Perform SR 3.2.1.1 and SR 3.2.1.2.~~	~~4 hours~~ ~~72 hours~~ INSERT 2 ~~72 hours~~ ~~Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits~~
DOC M03 ~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 2.	6 hours

4

4 **INSERT 2**

	CONDITION	REQUIRED ACTION		COMPLETION TIME
4.2.2.2.c.3 4.2.2.2.c.3.b 4.2.2.2.c.3	B. AFD min margin < 0	B.1	Reduce, by administrative means, positive AFD limit lines for each power level by $PSLOPE^{AFD}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	2 hours
		AND		
4.2.2.2.c.3.a Note *		B.2	Reduce, by administrative means, negative AFD limit lines for each power level by $NSLOPE^{AFD}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	2 hours
4.2.2.2.c.4 4.2.2.2.c.4.b Note **	C. $f_2(\Delta I)$ min margin < 0	C.1	Reduce Overpower ΔT positive $f_2(\Delta I)$ breakpoint limit by $PSLOPE^{f2(\Delta I)}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	48 hours
		AND		
4.2.2.2.c.4.a Note **		C.2	Reduce Overpower ΔT negative $f_2(\Delta I)$ breakpoint limit by $NSLOPE^{f2(\Delta I)}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	48 hours

CTS

F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ 1
X,Y,Z
3.2.1~~B~~ 2

Not required to be performed until 12 hours after

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------ 5

4.2.2.2 Note *** ~~During power escalation at the beginning of each cycle, THERMAL POWER may be increased until~~ an equilibrium power level has been achieved, at which a power distribution map ~~is~~ can be obtained. 5

4.2.2.1 DOC M04

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ is within the steady state limit. INSERT 3 6 1	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within ~~[~~12~~]~~ hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ was last verified 7 1 <u>AND</u> ~~[31 EFPD thereafter~~ ~~OR~~ 8 In accordance with the Surveillance Frequency Control Program ~~]~~ 8

4.2.2.2
4.2.2.3
DOC M06
DOC M09
DOC A02

4.2.2.2.d.1
DOC M07

4.2.2.2.d.2

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6 **INSERT 3**

---NOTE-----------------------------------
Not required to be performed if SR 3.2.1.2 and SR 3.2.1.3 are met.
4.2.2.2 --

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CTS

F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ X,Y,Z
3.2.1~~B~~

1 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.2.1.2~~	~~-----NOTE-----~~ ~~If measurements indicate that the~~ ~~maximum over z [~~ $F_Q^C(Z)$ ~~/ K(Z)]~~ ~~has increased since the previous evaluation of~~ $F_Q^C(Z)$ ~~:~~ ~~a. Increase~~ $F_Q^W(Z)$ ~~by the greater of a factor of [1.02] or by an appropriate factor specified in the COLR and reverify~~ $F_Q^W(Z)$ ~~is within limits or~~ ~~b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the~~ ~~maximum over z [~~ $F_Q^C(Z)$ ~~/ K(Z)]~~ ~~has not increased.~~ ~~----------~~	
	~~Verify~~ $F_Q^W(Z)$ ~~is within limit.~~	~~Once after each refueling prior to THERMAL POWER exceed-ing 75% RTP~~ ~~AND~~ ~~Once within [12] hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which~~ $F_Q^W(Z)$ ~~was last verified~~ ~~AND~~

4

CTS

F_Q(~~Z~~) X,Y,Z ~~(RAOC-W(Z) Methodology)~~ 1 2
3.2.1~~B~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
	~~[31 EFPD thereafter~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

4

INSERT 4

4

CTS 3.2.1

4 **INSERT 4**

SURVEILLANCE		FREQUENCY	
SR 3.2.1.2	------------------------------NOTE------------------------------ If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z) a. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify AFD min margin ~~>~~ ≥ 0; or b. Repeat SR 3.2.1.2 prior to the time at which the projected AFD min margin will be < 0. --		4.2.2.2.e 4.2.2.2.e.1 SII 4.2.2.2.e.2
	Verify AFD min margin ~~>~~ ≥ 0.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last verified <u>AND</u>	4.2.2.2.c.1 DOC M06 SII 4.2.2.2.d.1 DOC M07

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CTS 3.2.1

4 **INSERT 4 (continued)**

SURVEILLANCE	FREQUENCY
4.2.2.2.d.2	~~[31 EFPD thereafter]~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

CTS

3.2.1

4 **INSERT 4 (continued)**

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	------------------------------NOTE------------------------------ 4.2.2.2.e If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z) 4.2.2.2.e.1 a. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $f_2(\Delta I)$ min margin ~~>~~ ≥ 0; or SII 4.2.2.2.e.2 b. Repeat SR 3.2.1.3 prior to the time at which the projected $f_2(\Delta I)$ min margin will be < 0. --	
4.2.2.2.c.1 DOC M06	Verify $f_2(\Delta I)$ min margin ~~>~~ ≥ 0.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP SII AND 4.2.2.2.d.1 DOC M07 Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last verified AND

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4 **INSERT 4 (continued)**

SURVEILLANCE	FREQUENCY
4.2.2.2.d.2	~~[31 EFPD thereafter~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]

CTS

F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 1 2
3.2.1~~B~~

3.2 POWER DISTRIBUTION LIMITS

3.2.1~~B~~ Heat Flux Hot Channel Factor (F_Q(~~Z~~ X,Y,Z)) ~~(RAOC-W(Z) Methodology)~~ 2 1

3.2.2 LCO 3.2.1~~B~~ F_Q(~~Z~~ X,Y,Z)~~, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$,~~ shall be within the limits specified in the COLR. 2 1

Applicability APPLICABILITY: MODE 1.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a DOC M01 DOC M02	A. ------------NOTE------------ Required Action A.~~4~~ 5 shall be completed whenever this Condition is entered. -----------------------------------	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ exceeds limit. INSERT 1 AND	15 minutes after each ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ determination	3 1 3
DOC L01 ACTION a.2 DOC M02	~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ not within the steady state limit.	A.~~2~~ 4 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ exceeds limit. AND	72 hours after each ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ determination	3 1
ACTION b DOC M02		A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ exceeds limit. AND	~~72~~ 48 hours after each ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ determination	3 1
ACTION c DOC M02		A.~~4~~ 5 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3.	Prior to increasing THERMAL POWER above the limit of Required Action A.1	3 4

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3 **INSERT 1**

ACTION a.1
DOC M02

CONDITION	REQUIRED ACTION	COMPLETION TIME
	AND A.2 Reduce, by administrative means, AFD limits ≥ 1% for each 1% $F_Q^C(X,Y,Z)$ exceeds limit.	2 hours after each $F_Q^C(X,Y,Z)$ determination

CTS

F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ X,Y,Z
3.2.1~~B~~ 1 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. ------------NOTE------------ Required Action B.4 shall be completed whenever this Condition is entered. ---------------------------------~~ ~~$F_Q^W(Z)$ not within limits.~~	~~B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.~~ ~~AND~~ ~~B.2 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.~~ ~~AND~~ ~~B.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.~~ ~~AND~~ ~~B.4 Perform SR 3.2.1.1 and SR 3.2.1.2.~~	~~4 hours~~ ~~72 hours~~ INSERT 2 ~~72 hours~~ ~~Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits~~
~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 2.	6 hours

DOC M03

4

SEQUOYAH UNIT 2 ~~WOG STS~~

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CTS 3.2.1

4 **INSERT 2**

	CONDITION	REQUIRED ACTION		COMPLETION TIME
4.2.2.2.c.3 4.2.2.2.c.3.b 4.2.2.2.c.3	B. AFD min margin < 0	B.1	Reduce, by administrative means, positive AFD limit lines for each power level by $PSLOPE^{AFD}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	2 hours
		AND		
4.2.2.2.c.3.a Note *		B.2	Reduce, by administrative means, negative AFD limit lines for each power level by $NSLOPE^{AFD}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	2 hours
4.2.2.2.c.4 4.2.2.2.c.4.b Note **	C. $f_2(\Delta I)$ min margin < 0	C.1	Reduce Overpower ΔT positive $f_2(\Delta I)$ breakpoint limit by $PSLOPE^{f2(\Delta I)}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	48 hours
		AND		
4.2.2.2.c.4.a Note **		C.2	Reduce Overpower ΔT negative $f_2(\Delta I)$ breakpoint limit by $NSLOPE^{f2(\Delta I)}$ for each 1% $F_Q(X,Y,Z)$ exceeds limit.	48 hours

CTS

F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 1 2

3.2.1~~B~~

Not required to be performed until 12 hours after

SURVEILLANCE REQUIREMENTS

4.2.2.2 Note *** ---NOTE--- 5

~~During power escalation at the beginning of each cycle, THERMAL POWER may be increased until~~ an equilibrium power level has been achieved, at which a power distribution map ~~is~~ can be obtained. 5

4.2.2.1 DOC M04

SURVEILLANCE	FREQUENCY
4.2.2.2 4.2.2.3 DOC M06 DOC M09 DOC A02 SR 3.2.1.1 Verify ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ is within the steady state limit. INSERT 3 6 1	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 4.2.2.2.d.1 DOC M07 Once within ~~[~~12~~]~~ hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ was last verified 7 1 <u>AND</u> 4.2.2.2.d.2 ~~[31 EFPD thereafter~~ ~~OR~~ 8 In accordance with the Surveillance Frequency Control Program ~~]~~ 8

SEQUOYAH UNIT 2 ~~WOG STS~~

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6 **INSERT 3**

--NOTE-----------------------------------
Not required to be performed if SR 3.2.1.2 and SR 3.2.1.3 are met.
4.2.2.2 --

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CTS

F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ X,Y,Z 1
3.2.1~~B~~ 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.2.1.2~~	~~-----NOTE-----~~ ~~If measurements indicate that the~~ ~~maximum over z [~~ F_Q^C(Z) ~~/ K(Z)]~~ ~~has increased since the previous evaluation of~~ F_Q^C(Z)~~:~~ ~~a. Increase~~ F_Q^W(Z) ~~by the greater of a factor of [1.02] or by an appropriate factor specified in the COLR and reverify~~ F_Q^W(Z) ~~is within limits or~~ ~~b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the~~ ~~maximum over z [~~F_Q^C(Z) ~~/ K(Z)]~~ ~~has not increased.~~ ~~----------~~ ~~Verify~~ F_Q^W(Z) ~~is within limit.~~	~~Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~ ~~AND~~ ~~Once within [12] hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which~~ F_Q^W(Z) ~~was last verified~~ ~~AND~~

4

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CTS

F_Q(~~Z~~) X,Y,Z ~~(RAOC-W(Z) Methodology)~~ 1 2
3.2.1~~B~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
	~~[31 EFPD thereafter~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

4

INSERT 4 4

CTS 3.2.1

4 **INSERT 4**

CTS		SURVEILLANCE	FREQUENCY	
4.2.2.2.e 4.2.2.2.e.1 4.2.2.2.e.2	SR 3.2.1.2	-------------------------------NOTE------------------------------ If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z) a. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify AFD min margin > 0; or ≥ b. Repeat SR 3.2.1.2 prior to the time at which the projected AFD min margin will be < 0. ---		SII
4.2.2.2.c.1 DOC M06 4.2.2.2.d.1 DOC M07		Verify AFD min margin > 0. ≥	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last verified AND	SII

4 **INSERT 4 (continued)**

SURVEILLANCE	FREQUENCY
4.2.2.2.d.2	~~[31 EFPD thereafter]~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

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CTS

3.2.1

4 **INSERT 4 (continued)**

CTS		SURVEILLANCE	FREQUENCY	
4.2.2.2.e	SR 3.2.1.3	-------------------------------NOTE------------------------------ If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z)		
4.2.2.2.e.1		a. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $f_2(\Delta I)$ min margin ≥ 0; or ≥		SII
4.2.2.2.e.2		b. Repeat SR 3.2.1.3 prior to the time at which the projected $f_2(\Delta I)$ min margin will be < 0. ---		
4.2.2.2.c.1 DOC M06		Verify $f_2(\Delta I)$ min margin ≥ 0. ≥	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u>	SII
4.2.2.2.d.1 DOC M07			Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last verified <u>AND</u>	

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4 **INSERT 4 (continued)**

SURVEILLANCE	FREQUENCY
4.2.2.2.d.2	~~[31 EFPD thereafter~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

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JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The RAOC-W(Z) methodology and the Specification designator "B" are deleted because they are unnecessary. (Only one Heat Flux Hot Channel Factor Specification is used in the SQN ITS). This information is provided in NUREG-1431, Rev. 4 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the CAOC-F_{XY} and CAOC-W(Z) methodology Specifications (ISTS 3.2.1A and 3.2.1C) are not used and are not shown.

3. ISTS ACTIONS do not contain a requirement to reduce the AFD limits when ACTION A is entered for $F_Q^C(X,Y,Z)$ not met. CTS 3.2.2 ACTION a.1 requires a reduction of the allowable power at each point along the AFD limit lines to be reduced within 2 hours. This requirement and Completion Time are being added as Required Action A.2.

4. ISTS SR 3.2.1.2 and ISTS ACTION B have been deleted. CTS does not include requirements to verify $F_Q^W(Z)$ is within limits, or actions to take if $F_Q^W(Z)$ is not within limits. However, CTS does require the verification that both AFD min
≥ margin is > 0 and $f_2(\Delta I)$ min margin is > 0. Additionally, the CTS specifies the SII
actions to take if the above verifications are not met. These verifications and actions are added to ITS 3.2.1 as SR 3.2.1.2 and SR 3.2.1.3 with the associated ACTIONS B and C.

5. ISTS 3.2.1 Surveillance Requirements Note allows, during power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map is obtained. CTS 3.2.2 *** Note has a similar allowance. However, in both CTS and ISTS the allowance is for the first power escalation at the beginning of a new core cycle. Additionally, the CTS has SR 4.2.2.1 which provides, The provisions of Specification 4.0.4 are not applicable. This allowance enables SQN to enter the MODE of Applicability with the Surveillance not being met. ISTS does not have a similar allowance in LCO 3.2.1. Therefore, SQN is retaining the allowance to change the MODE of Applicability with the surveillance not being met by modifying the existing Surveillance Note.

6. ISTS 3.2.1.1 has been modified by a Note providing an allowance to not perform SR 3.2.1.1 if the Surveillance has been determined to be met based on the performance results of both SR 3.2.1.2 and SR 3.2.1.3. If both the AFD min
≥ 0 margin and the $f_2(\Delta I)$ min margin ~~are positive~~, then the steady state limit is met SII
because these margins represent bounding limiting conditions.

7. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

8. ISTS SR 3.2.1.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~B~~ Heat Flux Hot Channel Factor (F_Q(~~Z~~) X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 2 1

BASES

BACKGROUND

The purpose of the limits on the values of F_Q(~~Z~~ X,Y,Z) is to limit the local (i.e., pellet) peak power density. The value of F_Q(~~Z~~ X,Y,Z) varies along the axial height (Z) of the core and by assembly location, X, Y. 1

F_Q(~~Z~~ X,Y,Z) is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density~~, assuming nominal fuel pellet and fuel rod dimensions~~. Therefore, F_Q(~~Z~~ X,Y,Z) is a measure of the peak fuel pellet power within the reactor core. 8

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO(QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

F_Q(~~Z~~ X,Y,Z) varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. 1

F_Q(~~Z~~ X,Y,Z) is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions. 1

Using the measured three dimensional power distributions, it is possible to derive a measured value for F_Q(~~Z~~ X,Y,Z). However, because this value represents an equilibrium condition, it does not include the variations in the value of F_Q(~~Z~~ X,Y,Z) which are present during nonequilibrium situations such as load following or power ascension. 1

To account for these possible variations, INSERT 1 ~~the equilibrium value of F_Q(Z) is adjusted as F_Q^W(Z) by an elevation dependent factor that accounts for the calculated worst case transient conditions.~~

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

3.2.1

1 **INSERT 1**

"the F_Q(X,Y,Z) limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted by pre-calculated factors (MQ(X,Y,Z) and MC(X,Y,Z) respectively) to account for the calculated worst case transient conditions."

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F_Q(~~Z~~) X,Y,Z ~~(RAOC-W(Z) Methodology)~~ 1
B 3.2.1~~B~~ 2

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on F_Q(~~Z~~ X,Y,Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting. 1

F_Q(~~Z~~ X,Y,Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F_Q(~~Z~~ X,Y,Z) limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents . 8

F_Q(~~Z~~ X,Y,Z) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, F_Q(~~Z~~ X,Y,Z), shall be limited by the following relationships: 1

F_Q(~~Z~~ X,Y,Z) ≤ (~~CFQ~~ F_Q^{RTP} / P) K(Z) for P > 0.5

F_Q(~~Z~~ X,Y,Z) ≤ (~~CFQ~~ F_Q^{RTP} / 0.5) K(Z) for P ≤ 0.5

where: ~~CFQ~~ F_Q^{RTP} is the F_Q(~~Z~~ X,Y,Z) limit at RTP provided in the COLR,

K(Z) is the normalized ~~FQ(Z)~~ F_Q (X,Y,Z) as a function of core height provided in the COLR, and

P = THERMAL POWER / RTP

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F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ 1
X,Y,Z
B 3.2.1~~B~~ 2

BASES

LCO (continued)

SQN F_Q^{RTP}

F_Q^{RTP} For ~~this facility~~, the actual values of ~~CFQ~~ and K(Z) are given in the COLR; 2.62 1 3
however, ~~CFQ~~ is normally a number on the order of [2.32]~~, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1~~.

INSERT 2 ~~For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.~~

~~An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,~~

~~$F_Q^C(Z) = F_Q^M(Z)$ [1.0815]~~

~~where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.~~ 4

~~$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.~~

~~The expression for $F_Q^W(Z)$ is:~~

~~$F_Q^W(Z) = F_Q^C(Z)$ W(Z)~~

~~where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.~~

~~The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.~~

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation
X,Y,Z that it can stay within the LOCA F_Q(~~Z~~) limits. If ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ cannot be maintained within the LCO limits, reduction of the core power is required 1
~~and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required~~. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power. 5

X,Y,Z
X,Y,Z Violating the LCO limits for F_Q(~~Z~~) produces unacceptable consequences if a design basis event occurs while F_Q(~~Z~~) is outside its specified limits. 1

4 **INSERT 2**

Measured $F_Q(X,Y,Z)$ is compared against three limits:

- Steady state limit, $(F_Q^{RTP} / P) * K(Z)$,
- Limiting condition LOCA limit, BQDES(X,Y,Z), and
- Limiting condition centerline fuel melt limit, BCDES(X,Y,Z).

$F_Q(X,Y,Z)$ is approximated by $F_Q^C(X,Y,Z)$ for the steady state limit on $F_Q(X,Y,Z)$. An $F_Q^C(X,Y,Z)$ evaluation requires using the moveable incore detectors to obtain a power distribution map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(X,Y,Z)$) of $F_Q(X,Y,Z)$. Then,

$$F_Q^C(X,Y,Z) = \text{overall measured } F_Q(X,Y,Z) * 1.05 * 1.03$$

where, 1.05 is the measurement reliability factor that accounts for flux map measurement uncertainty (Reference 5) and 1.03 is the local engineering heat flux hot channel factor to account for fuel rod manufacturing tolerance (Reference 4).

BQDES(X,Y,Z) and BCDES(X,Y,Z) are cycle dependent design limits to ensure the $F_Q(X,Y,Z)$ limit is met during transients. An evaluation of these limits requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the assembly nodal measured value ($F_Q^M(X,Y,Z)$) of $F_Q(X,Y,Z)$. $F_Q^M(X,Y,Z)$ is compared directly to the limits BQDES(X,Y,Z) and BCDES(X,Y,Z). This is appropriate since BQDES(X,Y,Z) and BCDES(X,Y,Z) have been adjusted for uncertainties.

The expression for BQDES(X,Y,Z) is: BQDES(X,Y,Z) = $P^d(X,Y,Z)$ * MQ(X,Y,Z) * NRF * F1 / MRF

where:

- BQDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y at axial location Z. BQDES(X,Y,Z) ensures that the LOCA limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- $P^d(X,Y,Z)$ is the design power distribution for fuel assembly X,Y at axial location Z, including the operational flexibility factor;
- MQ(X,Y,Z) is the minimum available margin ratio for the LOCA limit at assembly X,Y and axial location Z;
- NRF is the nuclear reliability factor;
- F1 is the spacer grid factor;
- MRF is measurement reliability factor.

4

INSERT 2 (continued)

The expression for BCDES(X,Y,Z) is: BCDES(X,Y,Z) = P^d(X,Y,Z) * MC(X,Y,Z) * NRF * F1 / MRF

where:

- BCDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y, at axial location Z. BCDES(X,Y,Z) ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;

- MC(X,Y,Z) is the minimum available margin ratio for the centerline fuel melt limit at assembly X,Y and axial location Z;

The reactor core is operating as designed if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is operating as designed if the following relationship is satisfied:

$$F_Q^M(X,Y,Z) \leq \text{BQNOM}(X,Y,Z)$$

where:

- BQNOM(X,Y,Z) is the nominal design peaking factor for assembly X,Y at axial location Z increased by an allowance for the expected deviation between the measured and predicted design power distribution.

The F_Q(X,Y,Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in the COLR.

BASES

APPLICABILITY

X,Y,Z
The F_Q(~~Z~~) limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ exceeds its limit, maintains an acceptable absolute power density. ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ is ~~F_Q^M(Z)~~ $F_Q^M(X,Y,Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. ~~F_Q^M(Z)~~ $F_Q^M(X,Y,Z)$ is the measured value of ~~F_Q(Z).~~ F_Q (X,Y,Z) The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ and would require power reductions within 15 minutes of the ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit. 5

INSERT 3

A.~~2~~ 4

A reduction of the Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% by which ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.~~2~~ 4 may be affected by subsequent determinations of ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints. 5

Move to next page after A.3

3.2.1

5

INSERT 3

A.2

Required Action A.2 requires an administrative reduction of the AFD limits by ≥ 1% for each 1% by which $F_Q^C(X, Y, Z)$ exceeds the steady state limit. The allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded. The maximum allowable AFD limits initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(X, Y, Z)$ and would require further AFD limit reductions within 2 hours of the $F_Q^C(X, Y, Z)$ determination, if necessary to comply with the decreased maximum allowable AFD limits. Decreases in $F_Q^C(X, Y, Z)$ would allow increasing the maximum allowable AFD limits.

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F_Q(~~Z~~) X,Y,Z ~~(RAOC-W(Z) Methodology)~~ 1
B 3.2.1~~B~~ 2

BASES

ACTIONS (continued)

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥ 1% in ΔT span for each 1% by which ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of ~~72~~ 48 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ and would require Overpower ΔT trip setpoint reductions within ~~72~~ 48 hours of the ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints. 5

A.~~4~~ 5

Verification that ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ has been restored to within its steady state and transient limit, by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions. 1 5

INSERT 4

Condition A is modified by a Note that requires Required Action A.~~4~~ 5 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.~~4~~ 5. Performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 are necessary to assure ~~F_Q(Z)~~ $F_Q^C(X,Y,Z)$ is properly evaluated prior to increasing THERMAL POWER. 5 5 5 1 4 5

INSERT 5

INSERT 6

~~B.1~~

~~If it is found that the maximum calculated value of F_Q(Z) that can occur during normal maneuvers, F_Q^W(Z), exceeds its specified limits, there exists a potential for F_Q^C(Z) to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which F_Q^W(Z) exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.~~ 5

5 **INSERT 4**

Since $F_Q^C(X,Y,Z)$ exceeds the steady state limit, the limiting condition operational limit (BQDES) and the limiting condition Reactor Protection System limit (BCDES) may also be exceeded. By performing SR 3.2.1.2 and SR 3.2.1.3, appropriate actions with respect to reductions in AFD limits and OPΔT trip setpoints will be performed, ensuring that core conditions during operational and Condition II transients are maintained within the bounds of the safety analysis.

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3.2.1

4 5

INSERT 5

B.1 and B.2

The $F_Q(X,Y,Z)$ margin supporting AFD operational limits (AFD margin) during transient operations is based on the relationship between $F_Q^M(X,Y,Z)$ and the limiting condition operational limit, BQDES (X,Y,Z), as follows:

$$\%\text{AFD margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) * 100\%$$

The AFD min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then $F_Q^M(X,Y,Z)$ is less than BQDES (X,Y,Z) and calculation of %AFD margin is not required. If the AFD margin is less than zero, then $F_Q^M(X,Y,Z)$ is greater than BQDES (X,Y,Z) and the AFD limits may not be adequate to prevent exceeding the peaking criteria for a LOCA if a normal operational transient occurs.

Required Actions B.1 and B.2 require reducing the AFD limit lines as follows. The AFD limit reduction is from the full power AFD limits. The adjusted AFD limits must be used until a new measurement shows that a smaller adjustment can be made to the AFD limits, or that no adjustment is necessary:

APL = PAFDL – Absolute Value of ($PSLOPE^{AFD}$ * % AFD Margin)
ANL = NAFDL + Absolute Value of ($NSLOPE^{AFD}$ * % AFD Margin)

where,

- APL is the adjusted positive AFD limit.
- ANL is the adjusted negative AFD limit.
- PAFDL is the positive AFD limit defined in the COLR.
- NAFDL is the negative AFD limit defined in the COLR.
- $PSLOPE^{AFD}$ is the adjustment to the positive AFD limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BQDES (X,Y,Z) as defined in the COLR.
- $NSLOPE^{AFD}$ is the adjustment to the negative AFD limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BQDES (X,Y,Z) as defined in the COLR.
- % AFD Margin is the most negative margin determined above.

Completing Required Actions B.1 and B.2 within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

3.2.1

4 5 **INSERT 6**

C.1 and C.2

The $F_Q(X,Y,Z)$ margin supporting the Overpower ΔT $f_2(\Delta I)$ breakpoints ($f_2(\Delta I)$ margin) during transient operations is based on the relationship between $F_Q^M(X,Y,Z)$ and the limit, BCDES(X,Y,Z), as follows:

$$\% \ f_2(\Delta I) \text{ margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) * 100\%$$

The $f_2(\Delta I)$ min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then $F_Q^M(X,Y,Z)$ is less than BCDES(X,Y,Z) and calculation of % $f_2(\Delta I)$ margin is not required. If the $f_2(\Delta I)$ margin is less than zero, then $F_Q^M(X,Y,Z)$ is greater than BCDES(X,Y,Z) and there is a potential that the $f_2(\Delta I)$ limits are insufficient to preclude centerline fuel melt during certain transients.

Required Actions C.1 and C.2 require reducing the $f_2(\Delta I)$ breakpoint limits as follows. The $f_2(\Delta I)$ breakpoint limit reduction is always from the full power $f_2(\Delta I)$ breakpoint limits. The adjusted $f_2(\Delta I)$ breakpoint limits must be used until a new measurement shows that a smaller adjustment can be made to the $f_2(\Delta I)$ breakpoint limits, or that no adjustment is necessary.

$Posf_2(\Delta I)_{Adjusted}$ = $Posf_2(\Delta I)^{Limit}$ – Absolute Value of ($PSLOPE^{f2(\Delta I)}$ * % $f_2(\Delta I)$ Margin)
$Negf_2(\Delta I)_{Adjusted}$ = $Negf_2(\Delta I)^{Limit}$ + Absolute Value of ($NSLOPE^{f2(\Delta I)}$ * % $f_2(\Delta I)$ Margin)

where:

- $Posf_2(\Delta I)_{Adjusted}$ is the adjusted OPΔT positive $f_2(\Delta I)$ breakpoint limit.
- $Negf_2(\Delta I)_{Adjusted}$ is the adjusted OPΔT negative $f_2(\Delta I)$ breakpoint limit.
- $Posf_2(\Delta I)^{Limit}$ is the OPΔT positive $f_2(\Delta I)$ breakpoint limit defined in the COLR.
- $Negf_2(\Delta I)^{Limit}$ is the OPΔT negative $f_2(\Delta I)$ breakpoint limit defined in the COLR.
- $PSLOPE^{f2(\Delta I)}$ is the adjustment to the positive OPΔT $f_2(\Delta I)$ limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BCDES(X,Y,Z) as defined in the COLR.
- $NSLOPE^{f2(\Delta I)}$ is the adjustment to the negative OPΔT $f_2(\Delta I)$ limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BCDES(X,Y,Z) as defined in the COLR.
- % $f_2(\Delta I)$ Margin is the most negative margin determined above.

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3.2.1

4 5 **INSERT 6 (continued)**

Completing Required Actions C.1 and C.2 is a conservative action for protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Completing Required Actions C.1 and C.2 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

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BASES

ACTIONS (continued)

~~The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F^C_Q(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for Required Actions B.2, B.3 and B.4.~~

~~B.2~~

~~A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.~~

~~B.3~~ 5

~~Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.~~

~~B.4~~

~~Verification that $F^W_Q(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.~~

~~Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.~~

BASES

ACTIONS (continued)

~~C~~ D.1 5

If Required Actions A.1 through ~~A.4 or B.1 through B.4~~ A.5, B.1, B.2, C.1 or C.2 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 are modified by a Note. ~~The Note applies during the first power ascension after a refueling.~~ It states that Surveillance performance is not required ~~THERMAL POWER may be increased~~ until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained. 5

This allowance is modified, however, by one of the Frequency conditions that requires verification that ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q(~~Z~~ X,Y,Z) was last measured. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.1

~~Verification~~ Direct verification that ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ is within its specified limits involves increasing the overall measured F_Q (X,Y,Z) ~~$F^M_Q(Z)$~~ to allow for manufacturing tolerance and measurement uncertainties in order to obtain ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$. ~~Specifically, $F^M_Q(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F^C_Q(Z)$ = $F^M_Q(Z)$ [1.0815] (Ref. 4).~~ ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ is then compared to its specified limits. 1 3

The limit with which ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

INSERT 7 4

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased. 1

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$, another evaluation of this factor is required ~~[~~12~~]~~ hours after achieving equilibrium conditions at this higher power level (to ensure that ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits). 1 2 1

~~[The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 7

3.2.1

4 **INSERT 7**

The surveillance has been modified by a Note providing an allowance to not perform SR 3.2.1.1 if the Surveillance has been determined to be met based on the performance results of both SR 3.2.1.2 and SR 3.2.1.3. If both the AFD min margin and the $f_2(\Delta I)$ min margin ~~are positive~~ ≥ 0, then the steady state limit is met because these margins represent bounding limiting conditions. However, if AFD min margin or $f_2(\Delta I)$ min margin is negative then a direct evaluation of the steady state limit is required to satisfy this surveillance requirement. SII

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 and 3.2.1.3 5 2

The nuclear design process includes calculations performed to determine that the core can be operated within the ~~F_Q(Z)~~ F_Q (X,Y,Z) limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of INSERT 8 ~~core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum F_Q(Z) calculated to occur in normal operation, $F_Q^W(Z)$~~.

INSERT 9 ~~The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR~~.

~~The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations~~ BQDES (X,Y,Z) and BCDES (X,Y,Z) limits are not applicable for the following axial core regions, measured in percent of core height: 4

a. Lower core region, from 0 to 15% inclusive and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed based on future projections. If ~~$F_Q^W(Z)$~~ $F_Q^M(X,Y,Z)$ is evaluated and found to be within the applicable limiting condition limits, an evaluation ~~of the expression below~~ is required to account for any increase to ~~$F_Q^M(Z)$~~ $F_Q^M(X,Y,Z)$ that may occur and cause the F_Q~~(Z)~~ F_Q (X,Y,Z) limit to be exceeded before the next required ~~F_Q(Z)~~ F_Q (X,Y,Z) evaluation.

INSERT 10

~~If the two most recent F_Q(Z) evaluations show an increase in the expression maximum over z [$F_Q^C(Z)$ / K(Z)], it is required to meet the F_Q(Z) limit with the last $F_Q^W(Z)$ increased by the greater of a factor of [1.02] or by an appropriate factor specified in the COLR (Ref. 5)~~ 4

3.2.1

4 **INSERT 8**

both assembly and axial location (X,Y,Z), has been included in the cycle specific limits BQDES(X,Y,Z) and BCDES(X,Y,Z) using margin factors MQ(X,Y,Z) and MC(X,Y,Z), respectively (Reference 5).

4 **INSERT 9**

No uncertainties are applied to $F_Q^M(X,Y,Z)$ because the limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted for uncertainties.

4 **INSERT 10**

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in $F_Q^M(X,Y,Z)$ for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z), further consideration is required.

The implications of these extrapolations are considered separately for both the operational and RPS heat flux hot channel factor limits. If the extrapolations of $F_Q^M(X,Y,Z)$ are unfavorable, additional actions must be taken. These actions are to meet the F_Q(X,Y,Z) limit with the last $F_Q^M(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q^M(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements prevent F_Q(X,Y,Z) from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

F_Q(X,Y,Z) is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification within 12 hours after achieving equilibrium conditions to ensure that F_Q(X,Y,Z) is within its limit at higher power levels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification," February 1994, or other appropriate plant specific methodology, is to be listed in the COLR description in the Administrative Controls Section 5.0 to address the methodology used to derive this factor.~~
~~---~~

~~or to evaluate F_Q(Z) more frequently, each 7 EFPD. These alternative requirements prevent F_Q(Z) from exceeding its limit for any significant period of time without detection.~~ 4

~~Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the F_Q(Z) limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.~~

~~F_Q(Z) is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, [12] hours after achieving equilibrium conditions to ensure that F_Q(Z) is within its limit at higher power levels.~~

~~The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of F_Q(Z) evaluations.~~

~~[The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~--]~~

BASES

REFERENCES

1. 10 CFR 50.46, 1974.

2. Regulatory Guide 1.77, Rev. 0, May 1974.

3. 10 CFR 50, Appendix A, GDC 26.

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. ~~WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.~~ 4

BAW-10163PA “Core Operating Limit Methodology for Westinghouse-Designed PWRs” June 1989.

4

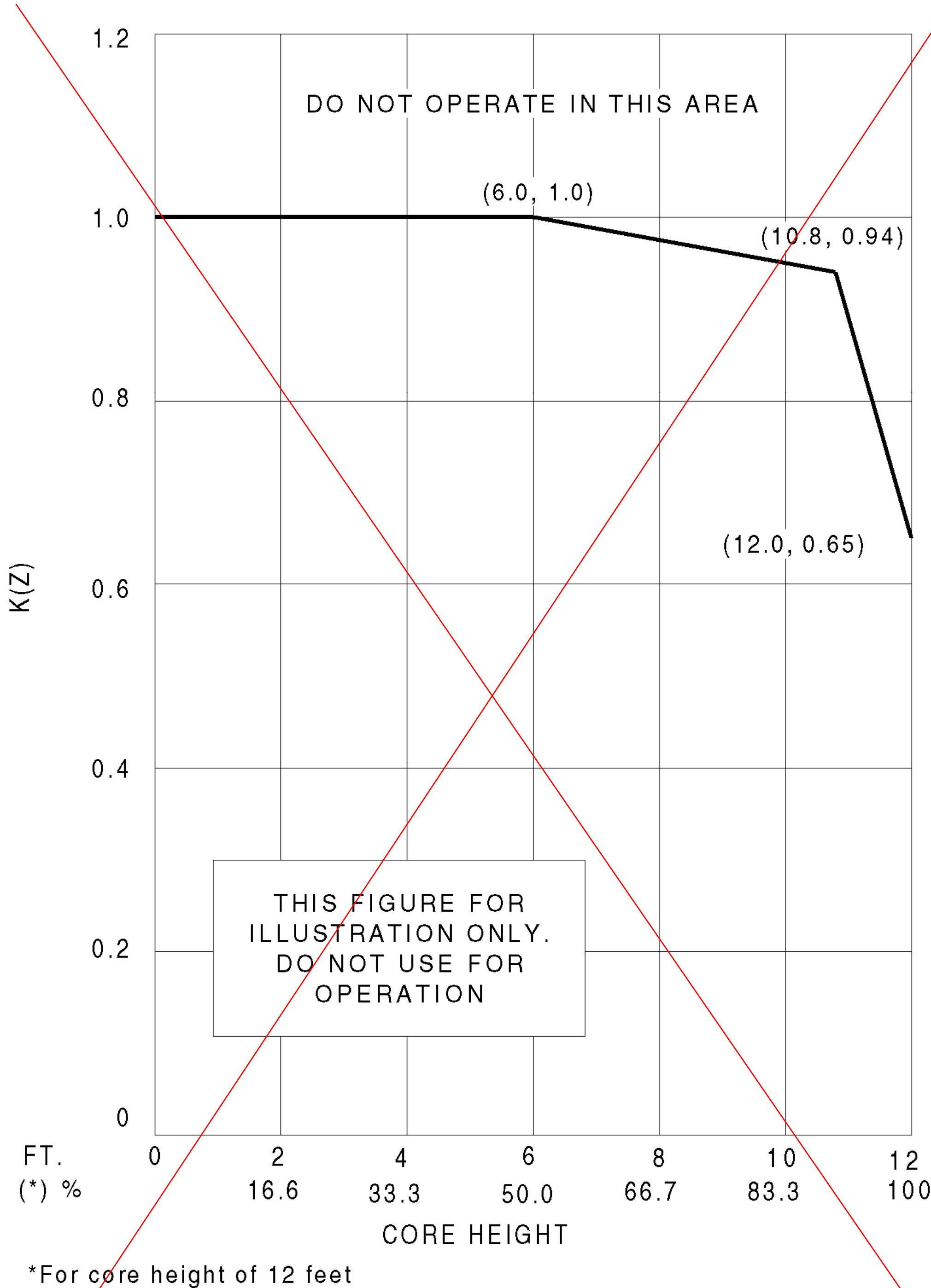

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized F_Q(Z) as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~B~~ Heat Flux Hot Channel Factor (F_Q(~~Z~~ X,Y,Z) ~~(RAOC-W(Z) Methodology)~~ 2 1

BASES

BACKGROUND

The purpose of the limits on the values of F_Q(~~Z~~ X,Y,Z) is to limit the local (i.e., pellet) peak power density. The value of F_Q(~~Z~~ X,Y,Z) varies along the axial height (Z) of the core and by assembly location, X, Y. 1

F_Q(~~Z~~ X,Y,Z) is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density~~, assuming nominal fuel pellet and fuel rod dimensions~~. Therefore, F_Q(~~Z~~ X,Y,Z) is a measure of the peak fuel pellet power within the reactor core. 8

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO(QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

F_Q(~~Z~~ X,Y,Z) varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. 1

F_Q(~~Z~~ X,Y,Z) is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions. 1

Using the measured three dimensional power distributions, it is possible to derive a measured value for F_Q(~~Z~~ X,Y,Z). However, because this value represents an equilibrium condition, it does not include the variations in the value of F_Q(~~Z~~ X,Y,Z) which are present during nonequilibrium situations such as load following or power ascension. 1

To account for these possible variations, INSERT 1 ~~the equilibrium value of F_Q(Z) is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions.~~

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

3.2.1

1 **INSERT 1**

"the $F_Q(X,Y,Z)$ limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted by pre-calculated factors (MQ(X,Y,Z) and MC(X,Y,Z) respectively) to account for the calculated worst case transient conditions."

F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ X,Y,Z 1
B 3.2.1~~B~~ 2

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on F_Q(~~Z~~ X,Y,Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

1

F_Q(~~Z~~ X,Y,Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F_Q(~~Z~~ X,Y,Z) limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents .

8

F_Q(~~Z~~ X,Y,Z) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, F_Q(~~Z~~ X,Y,Z), shall be limited by the following relationships:

F_Q(~~Z~~ X,Y,Z) ≤ (~~CFQ~~ F_Q^{RTP} / P) K(Z) for P > 0.5

F_Q(~~Z~~ X,Y,Z) ≤ (~~CFQ~~ F_Q^{RTP} / 0.5) K(Z) for P ≤ 0.5

1

where: ~~CFQ~~ F_Q^{RTP} is the F_Q(~~Z~~ X,Y,Z) limit at RTP provided in the COLR,

K(Z) is the normalized ~~FQ(Z)~~ F_Q (X,Y,Z) as a function of core height provided in the COLR, and

P = THERMAL POWER / RTP

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F_Q(~~Z~~) ~~(RAOC-W(Z) Methodology)~~ X,Y,Z
B 3.2.1~~B~~ 1 2

BASES

LCO (continued)

SQN F_Q^{RTP}
For ~~this facility~~, the actual values of ~~CFQ~~ and K(Z) are given in the COLR; 2.62 1 3
F_Q^{RTP} however, ~~CFQ~~ is normally a number on the order of [2.32]~~, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1~~.

INSERT 2 ~~For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.~~

~~An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,~~

~~$F_Q^C(Z) = F_Q^M(Z)$ [1.0815]~~

~~where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.~~ 4

~~$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.~~

~~The expression for $F_Q^W(Z)$ is:~~

~~$F_Q^W(Z) = F_Q^C(Z)$ W(Z)~~

~~where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.~~

~~The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.~~

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation X,Y,Z that it can stay within the LOCA F_Q(~~Z~~) limits. If ~~$F_Q^C(Z)$~~ $F_Q^C(X,Y,Z)$ cannot be maintained within the LCO limits, reduction of the core power is required ~~and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required~~. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power. 1 5

X,Y,Z X,Y,Z Violating the LCO limits for F_Q(~~Z~~) produces unacceptable consequences if a design basis event occurs while F_Q(~~Z~~) is outside its specified limits. 1

3.2.1

4 **INSERT 2**

Measured $F_Q(X,Y,Z)$ is compared against three limits:

- Steady state limit, $(F_Q^{RTP} / P) * K(Z)$,
- Limiting condition LOCA limit, BQDES(X,Y,Z), and
- Limiting condition centerline fuel melt limit, BCDES(X,Y,Z).

$F_Q(X,Y,Z)$ is approximated by $F_Q^C(X,Y,Z)$ for the steady state limit on $F_Q(X,Y,Z)$. An $F_Q^C(X,Y,Z)$ evaluation requires using the moveable incore detectors to obtain a power distribution map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(X,Y,Z)$) of $F_Q(X,Y,Z)$. Then,

$$F_Q^C(X,Y,Z) = \text{overall measured } F_Q(X,Y,Z) * 1.05 * 1.03$$

where, 1.05 is the measurement reliability factor that accounts for flux map measurement uncertainty (Reference 5) and 1.03 is the local engineering heat flux hot channel factor to account for fuel rod manufacturing tolerance (Reference 4).

BQDES(X,Y,Z) and BCDES(X,Y,Z) are cycle dependent design limits to ensure the $F_Q(X,Y,Z)$ limit is met during transients. An evaluation of these limits requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the assembly nodal measured value ($F_Q^M(X,Y,Z)$) of $F_Q(X,Y,Z)$. $F_Q^M(X,Y,Z)$ is compared directly to the limits BQDES(X,Y,Z) and BCDES(X,Y,Z). This is appropriate since BQDES(X,Y,Z) and BCDES(X,Y,Z) have been adjusted for uncertainties.

The expression for BQDES(X,Y,Z) is: BQDES(X,Y,Z) = $P^d(X,Y,Z)$ * MQ(X,Y,Z) * NRF * F1 / MRF

where:

- BQDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y at axial location Z. BQDES(X,Y,Z) ensures that the LOCA limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- $P^d(X,Y,Z)$ is the design power distribution for fuel assembly X,Y at axial location Z, including the operational flexibility factor;
- MQ(X,Y,Z) is the minimum available margin ratio for the LOCA limit at assembly X,Y and axial location Z;
- NRF is the nuclear reliability factor;
- F1 is the spacer grid factor;
- MRF is measurement reliability factor.

4

INSERT 2 (continued)

The expression for BCDES(X,Y,Z) is: BCDES(X,Y,Z) = P^d(X,Y,Z) * MC(X,Y,Z) * NRF * F1 / MRF

where:

- BCDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y, at axial location Z. BCDES(X,Y,Z) ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;

- MC(X,Y,Z) is the minimum available margin ratio for the centerline fuel melt limit at assembly X,Y and axial location Z;

The reactor core is operating as designed if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is operating as designed if the following relationship is satisfied:

$$F_Q^M(X,Y,Z) \leq \text{BQNOM}(X,Y,Z)$$

where:

- BQNOM(X,Y,Z) is the nominal design peaking factor for assembly X,Y at axial location Z increased by an allowance for the expected deviation between the measured and predicted design power distribution.

The F_Q(X,Y,Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in the COLR.

BASES

APPLICABILITY

X,Y,Z

The F_Q(~~Z~~) limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. 1

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ exceeds its limit, maintains an acceptable absolute power density. ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ is ~~F_Q^M(Z)~~ $F_Q^M(X,Y,Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. ~~F_Q^M(Z)~~ $F_Q^M(X,Y,Z)$ is the measured value of ~~F_Q(Z).~~ F_Q (X,Y,Z) The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ and would require power reductions within 15 minutes of the ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit. 5

INSERT 3

5

A.~~2~~ 4

A reduction of the Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% by which ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.~~2~~ 4 may be affected by subsequent determinations of ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in ~~F_Q^C(Z)~~ $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints. 5

Move to next page after A.3

5

INSERT 3

A.2

Required Action A.2 requires an administrative reduction of the AFD limits by ≥ 1% for each 1% by which $F_Q^C(X,Y,Z)$ exceeds the steady state limit. The allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded. The maximum allowable AFD limits initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(X,Y,Z)$ and would require further AFD limit reductions within 2 hours of the $F_Q^C(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable AFD limits. Decreases in $F_Q^C(X,Y,Z)$ would allow increasing the maximum allowable AFD limits.

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F_Q(~~Z~~) X,Y,Z ~~(RAOC-W(Z) Methodology)~~ 1
B 3.2.1~~B~~ 2

BASES

ACTIONS (continued)

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥ 1% in ΔT span for each 1% by which ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of ~~72~~ 48 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and would require Overpower ΔT trip setpoint reductions within ~~72~~ 48 hours of the ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints. 5

A.~~4~~ 5

Verification that ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ has been restored to within its steady state and transient limit, by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions. 1 5

INSERT 4

Condition A is modified by a Note that requires Required Action A.~~4~~ 5 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.~~4~~ 5. Performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 are necessary to assure ~~$F_Q(Z)$~~ $F^C_Q(X,Y,Z)$ is properly evaluated prior to increasing THERMAL POWER. 5 5 5 1

INSERT 5 4 5

INSERT 6

~~B.1~~

~~If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F^W_Q(Z)$, exceeds its specified limits, there exists a potential for $F^C_Q(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which $F^W_Q(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.~~ 5

3.2.1

5 **INSERT 4**

Since $F_Q^C(X, Y, Z)$ exceeds the steady state limit, the limiting condition operational limit (BQDES) and the limiting condition Reactor Protection System limit (BCDES) may also be exceeded. By performing SR 3.2.1.2 and SR 3.2.1.3, appropriate actions with respect to reductions in AFD limits and OPΔT trip setpoints will be performed, ensuring that core conditions during operational and Condition II transients are maintained within the bounds of the safety analysis.

3.2.1

4 5

INSERT 5

B.1 and B.2

The $F_Q(X,Y,Z)$ margin supporting AFD operational limits (AFD margin) during transient operations is based on the relationship between $F_Q^M(X,Y,Z)$ and the limiting condition operational limit, BQDES (X,Y,Z), as follows:

$$\%\text{AFD margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) * 100\%$$

The AFD min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then $F_Q^M(X,Y,Z)$ is less than BQDES (X,Y,Z) and calculation of %AFD margin is not required. If the AFD margin is less than zero, then $F_Q^M(X,Y,Z)$ is greater than BQDES (X,Y,Z) and the AFD limits may not be adequate to prevent exceeding the peaking criteria for a LOCA if a normal operational transient occurs.

Required Actions B.1 and B.2 require reducing the AFD limit lines as follows. The AFD limit reduction is from the full power AFD limits. The adjusted AFD limits must be used until a new measurement shows that a smaller adjustment can be made to the AFD limits, or that no adjustment is necessary:

APL = PAFDL – Absolute Value of ($PSLOPE^{AFD}$ * % AFD Margin)
ANL = NAFDL + Absolute Value of ($NSLOPE^{AFD}$ * % AFD Margin)

where,

- APL is the adjusted positive AFD limit.
- ANL is the adjusted negative AFD limit.
- PAFDL is the positive AFD limit defined in the COLR.
- NAFDL is the negative AFD limit defined in the COLR.
- $PSLOPE^{AFD}$ is the adjustment to the positive AFD limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BQDES (X,Y,Z) as defined in the COLR.
- $NSLOPE^{AFD}$ is the adjustment to the negative AFD limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BQDES (X,Y,Z) as defined in the COLR.
- % AFD Margin is the most negative margin determined above.

Completing Required Actions B.1 and B.2 within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

3.2.1

4 5 **INSERT 6**

C.1 and C.2

The $F_Q(X,Y,Z)$ margin supporting the Overpower ΔT $f_2(\Delta I)$ breakpoints ($f_2(\Delta I)$ margin) during transient operations is based on the relationship between $F_Q^M(X,Y,Z)$ and the limit, BCDES(X,Y,Z), as follows:

$$\% \ f_2(\Delta I) \text{ margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) * 100\%$$

The $f_2(\Delta I)$ min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then $F_Q^M(X,Y,Z)$ is less than BCDES(X,Y,Z) and calculation of % $f_2(\Delta I)$ margin is not required. If the $f_2(\Delta I)$ margin is less than zero, then $F_Q^M(X,Y,Z)$ is greater than BCDES(X,Y,Z) and there is a potential that the $f_2(\Delta I)$ limits are insufficient to preclude centerline fuel melt during certain transients.

Required Actions C.1 and C.2 require reducing the $f_2(\Delta I)$ breakpoint limits as follows. The $f_2(\Delta I)$ breakpoint limit reduction is always from the full power $f_2(\Delta I)$ breakpoint limits. The adjusted $f_2(\Delta I)$ breakpoint limits must be used until a new measurement shows that a smaller adjustment can be made to the $f_2(\Delta I)$ breakpoint limits, or that no adjustment is necessary.

$Posf_2(\Delta I)_{Adjusted}$ = $Posf_2(\Delta I)^{Limit}$ – Absolute Value of ($PSLOPE^{f2(\Delta I)}$ * % $f_2(\Delta I)$ Margin)
$Negf_2(\Delta I)_{Adjusted}$ = $Negf_2(\Delta I)^{Limit}$ + Absolute Value of ($NSLOPE^{f2(\Delta I)}$ * % $f_2(\Delta I)$ Margin)

where:

- $Posf_2(\Delta I)_{Adjusted}$ is the adjusted OPΔT positive $f_2(\Delta I)$ breakpoint limit.
- $Negf_2(\Delta I)_{Adjusted}$ is the adjusted OPΔT negative $f_2(\Delta I)$ breakpoint limit.
- $Posf_2(\Delta I)^{Limit}$ is the OPΔT positive $f_2(\Delta I)$ breakpoint limit defined in the COLR.
- $Negf_2(\Delta I)^{Limit}$ is the OPΔT negative $f_2(\Delta I)$ breakpoint limit defined in the COLR.
- $PSLOPE^{f2(\Delta I)}$ is the adjustment to the positive OPΔT $f_2(\Delta I)$ limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BCDES(X,Y,Z) as defined in the COLR.
- $NSLOPE^{f2(\Delta I)}$ is the adjustment to the negative OPΔT $f_2(\Delta I)$ limit required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds BCDES(X,Y,Z) as defined in the COLR.
- % $f_2(\Delta I)$ Margin is the most negative margin determined above.

3.2.1

4 5 **INSERT 6 (continued)**

Completing Required Actions C.1 and C.2 is a conservative action for protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Completing Required Actions C.1 and C.2 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

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BASES

ACTIONS (continued)

~~The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for Required Actions B.2, B.3 and B.4.~~

~~B.2~~

~~A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.~~

~~B.3~~

~~Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.~~

~~B.4~~

~~Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.~~

~~Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.~~

5

BASES

ACTIONS (continued)

~~C~~ D.1 5

If Required Actions A.1 through ~~A.4 or B.1 through B.4~~ A.5, B.1, B.2, C.1 or C.2 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.~~2~~ 3 are modified by a Note. ~~The Note applies during the first power ascension after a refueling.~~ It states that Surveillance performance is not required ~~THERMAL POWER may be increased~~ until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained. 5

This allowance is modified, however, by one of the Frequency conditions that requires verification that ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of ~~$F^C_Q(Z)$~~ $F^C_Q(X,Y,Z)$ and ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q(~~Z~~ X,Y,Z) was last measured. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.1

Direct verification ~~Verification~~ that $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~ is within its specified limits involves increasing the overall measured F_Q (X,Y,Z) ~~$F^M_Q(Z)$~~ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~. ~~Specifically, $F^M_Q(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F^C_Q(Z)$ = $F^M_Q(Z)$ [1.0815] (Ref. 4).~~ $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~ is then compared to its specified limits. 1 3

The limit with which $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

INSERT 7 4

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased. 1

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~, another evaluation of this factor is required ~~[~~12~~]~~ hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(X,Y,Z)$ ~~$F^C_Q(Z)$~~ values are being reduced sufficiently with power increase to stay within the LCO limits). 1 2 1

~~[The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).~~

~~OR~~ 6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 7

3.2.1

4 **INSERT 7**

≥ 0

SII

The surveillance has been modified by a Note providing an allowance to not perform SR 3.2.1.1 if the Surveillance has been determined to be met based on the performance results of both SR 3.2.1.2 and SR 3.2.1.3. If both the AFD min margin and the $f_2(\Delta I)$ min margin ~~are positive~~, then the steady state limit is met because these margins represent bounding limiting conditions. However, if AFD min margin or $f_2(\Delta I)$ min margin is negative then a direct evaluation of the steady state limit is required to satisfy this surveillance requirement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2
and 3.2.1.3
5
2

The nuclear design process includes calculations performed to determine that the core can be operated within the ~~F_Q(Z)~~ F_Q (X,Y,Z) limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of INSERT 8 ~~core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F^C_Q(Z)$, by W(Z) gives the maximum F_Q(Z) calculated to occur in normal operation, $F^W_Q(Z)$~~.

INSERT 9 ~~The limit with which $F^W_Q(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR~~.

~~The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F^W_Q(Z)$ evaluations~~ BQDES (X,Y,Z) and BCDES (X,Y,Z) limits are not applicable for the following axial core regions, measured in percent of core height:

4

a. Lower core region, from 0 to 15% inclusive and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed based on future projections. If ~~$F^W_Q(Z)$~~ $F^M_Q(X,Y,Z)$ is evaluated and found to be within the applicable limiting condition limits, an evaluation ~~of the expression below~~ is required to account for any increase to ~~$F^M_Q(Z)$~~ that may occur and cause the F_Q(~~Z~~) F_Q (X,Y,Z) limit to be exceeded before the next required ~~F_Q(Z)~~ evaluation.

INSERT 10

~~If the two most recent F_Q(Z) evaluations show an increase in the expression maximum over z [$F^C_Q(Z)$ / K(Z)], it is required to meet the F_Q(Z) limit with the last $F^W_Q(Z)$ increased by the greater of a factor of [1.02] or by an appropriate factor specified in the COLR (Ref. 5)~~

4

4 **INSERT 8**

both assembly and axial location (X,Y,Z), has been included in the cycle specific limits BQDES(X,Y,Z) and BCDES(X,Y,Z) using margin factors MQ(X,Y,Z) and MC(X,Y,Z), respectively (Reference 5).

4 **INSERT 9**

No uncertainties are applied to $F_Q^M(X,Y,Z)$ because the limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted for uncertainties.

4 **INSERT 10**

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in $F_Q^M(X,Y,Z)$ for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an $F_Q^M(X,Y,Z)$ > BQNOM(X,Y,Z), further consideration is required.

The implications of these extrapolations are considered separately for both the operational and RPS heat flux hot channel factor limits. If the extrapolations of $F_Q^M(X,Y,Z)$ are unfavorable, additional actions must be taken. These actions are to meet the F_Q(X,Y,Z) limit with the last $F_Q^M(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q^M(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements prevent F_Q(X,Y,Z) from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

F_Q(X,Y,Z) is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification within 12 hours after achieving equilibrium conditions to ensure that F_Q(X,Y,Z) is within its limit at higher power levels.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification," February 1994, or other appropriate plant specific methodology, is to be listed in the COLR description in the Administrative Controls Section 5.0 to address the methodology used to derive this factor.~~
~~--~~

~~or to evaluate F_Q(Z) more frequently, each 7 EFPD. These alternative requirements prevent F_Q(Z) from exceeding its limit for any significant period of time without detection.~~ 4

~~Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the F_Q(Z) limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.~~

~~F_Q(Z) is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, [12] hours after achieving equilibrium conditions to ensure that F_Q(Z) is within its limit at higher power levels.~~

~~The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of F_Q(Z) evaluations.~~

~~[The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~--]~~

BASES

REFERENCES	1. 10 CFR 50.46, 1974.
	2. Regulatory Guide 1.77, Rev. 0, May 1974.
	3. 10 CFR 50, Appendix A, GDC 26.
	4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
	5. ~~WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.~~ BAW-10163PA "Core Operating Limit Methodology for Westinghouse-Designed PWRs" June 1989. 4

F_Q(~~Z~~X,Y,Z) ~~(RAOC-W(Z) Methodology)~~
B 3.2.1~~B~~

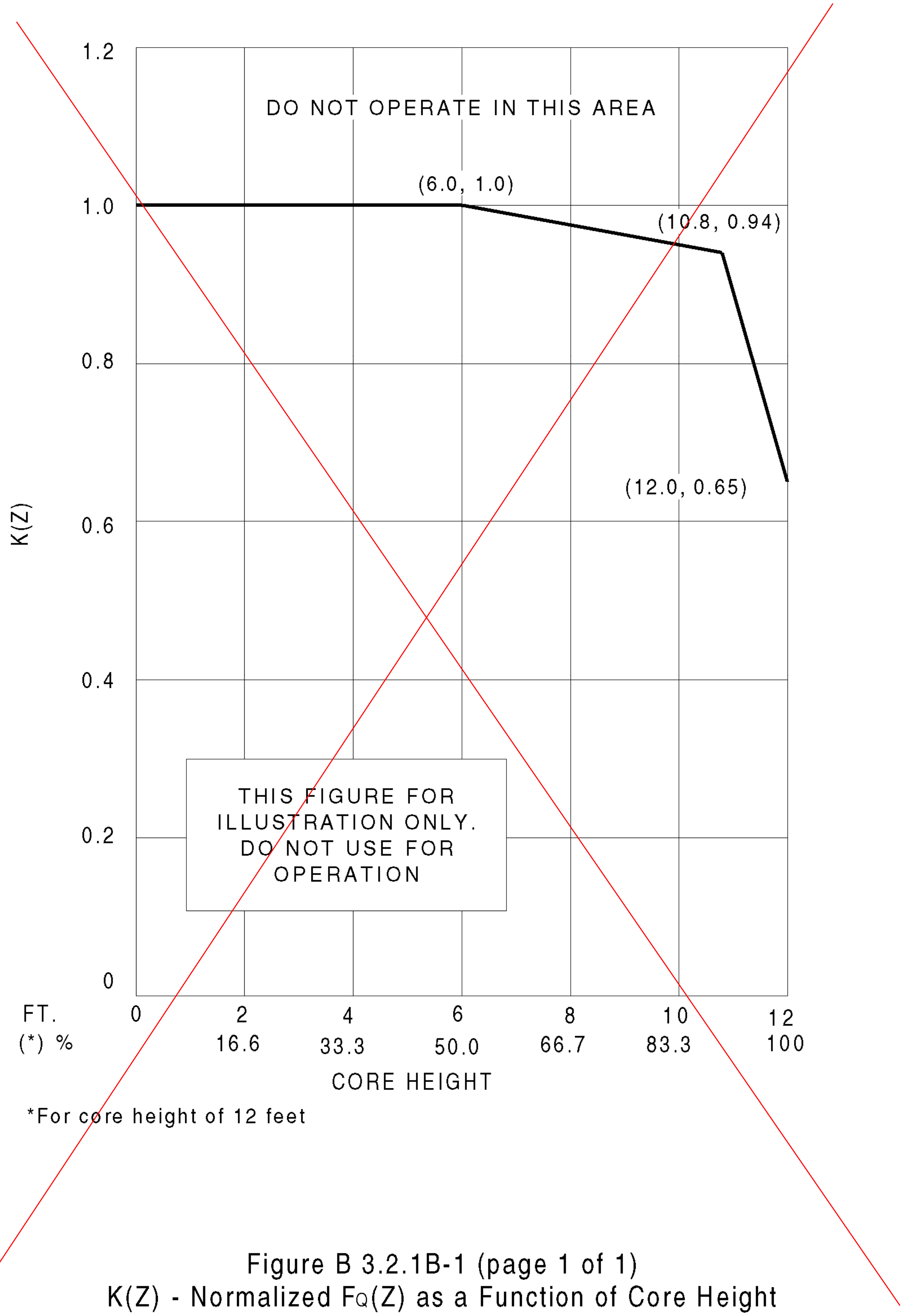

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized F_Q(Z) as a Function of Core Height

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, BASES, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The RAOC-W(Z) methodology and the Specification designator "B" are deleted because they are unnecessary. (Only one Heat Flux Hot Channel Factor Specification is used in the SQN ITS.) This information is provided in NUREG-1431, Rev. 4 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the CAOC-F_{XY} and CAOC-W(Z) methodology Specification Bases (ISTS B 3.2.1A and B 3.2.1C) are not used and are not shown.

3. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

4. The ISTS Bases for LCO 3.2.1, has been updated to reflect the methodology identified in BAW-10163PA “Core Operating Limit Methodology for Westinghouse-Designed PWRs” June 1989.

5. Changes have been made to be consistent with changes made to the Specification.

6. ISTS SR 3.2.1.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.2.1.1 under the Surveillance Frequency Control Program.

7. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

8. Editorial changes made to enhance clarity/consistency.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.1, HEAT FLUX HOT CHANNEL FACTOR ($F_Q(X,Y,Z)$)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}(X,Y)$)

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

A01

ITS ITS 3.2.2

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 $F_{\Delta H}(X,Y)$ shall be maintained within the limits specified in the COLR.

Applicability APPLICABILITY: MODE 1

ACTION:

Add proposed ACTION A Note M01

ACTION A With $F_{\Delta H}(X,Y)$ exceeding the limit specified in the COLR:

a. Within 2 hours either:

~~1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

Required Action A.1 2. Reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH~~*~~% for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and LA01

ACTION A b. Within the next ~~4~~ 72 hours either: L01

~~1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

LA01

Required Action A.2 2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH~~*~~% for each 1% that $F_{\Delta H}(X,Y)$ exceeds that limit, and

ACTION A c. Within 24 hours of initially being outside the limit specified in the COLR, either:

1. ~~Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

LA02

Required Action A.3 2. Verify ~~through incore flux mapping~~ that $F_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a.2 or

ACTION C reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours. L02

Add proposed ACTION C M02

~~* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA01

A01

ITS

ITS 3.2.2

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

Required Action A.4

d. Within 48 hours of initially being outside the limit specified in the COLR, reduce the Overtemperature Delta T ~~K_1 term in Table 2.2-1~~ by at least TRH~~**~~ for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and LA03

Add proposed Required Action A.5 Note A03

Required Action A.5

e. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b. and/or c. and/or d., above: subsequent POWER OPERATION may proceed provided that $F_{\Delta H}(X,Y)$ is demonstrated, ~~through incore flux mapping~~, to be within the above limit prior to exceeding the following THERMAL POWER levels: LA02

Completion Time A.5

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

~~** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA03

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

M03

SR NOTE 4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

SR 3.2.2.1
SR 3.2.2.2 4.2.3.2 $F_{\Delta H}^{M}(X,Y)$ shall be evaluated to determine if $F_{\Delta H}(X,Y)$ is within its limit by:

~~a. Using the movable incore detectors to obtain a power distribution map $F_{\Delta H}^{M}(X,Y)$ * at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Satisfying the following relationship:~~

~~$F\Delta HR^{M}(X,Y) \leq BHNOM(X,Y)$~~

~~Where:~~

$$\cancel{F\Delta HR^{M}(X,Y) = \frac{F_{\Delta H}^{M}(X,Y)}{MAP^{M} \;/\; AXIAL(X,Y)}}$$

~~And BHNOM(X,Y)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.~~

~~MAP^{M} is the maximum Allowable Peak** obtained from the measured power distribution.~~

~~AXIAL(X,Y) is the axial shape for $F_{\Delta H}(X,Y)$.~~

LA04

c. If the above relationship is not satisfied, then

1. For the location, calculate the % margin to the maximum allowable design as follows:

SR 3.2.2.1

$$\% \; F_{\Delta H} \text{ Margin } = \cancel{\left(1 - \frac{F\Delta HR^{M}(X,Y)}{BHDES(X,Y)}\right) x\ 100\%}$$

SR 3.2.2.2

$$\% \; f_{1}(\Delta I) Margin \; = \cancel{\left(1 - \frac{F\Delta HR^{M}(X,Y)}{BRDES(X,Y)}\right) x\ 100\%}$$

LA04

~~where BHDES (X,Y) and BRDES (X,Y)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within the LCO limits, and include allowances for calculational and measurement uncertainties.~~

~~* No additional uncertainties are required in the following equations for $F_{\Delta H}^{M}(X,Y)$ and $F)HR^{M}(X,Y)$, because the limits include uncertainties.~~

~~** BHNOM(X,Y), MAP^{M}, BHDES(X,Y), and BRDES(X,Y) data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.~~

LA04

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

~~2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above.~~

~~$F\Delta_H$ min margin = minimum % margin value of all locations examined~~

~~$f_1(\Delta I)$ min margin = minimum % margin value of all locations examined~~ LA04

ACTION A

3. If the $F_{\Delta H}$ min margin in 4.2.3.2.c.2 above is < 0, then within 2 hours reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH~~*~~% x most negative margin from 4.2.3.2.c.2 ~~and maintain the requirements of Specification 3.2.3; otherwise the Action statements for 3.2.3 apply.~~ LA01 M04

ACTION B

4. If the $f_1(\Delta I)$ min margin in 4.2.3.2.c.2 above is < 0, then within 48 hours reduce the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH~~**~~% x most negative margin from 4.2.3.2.c.2 ~~and maintain the requirements of Specification 3.2.3; otherwise the action statements for 3.2.3 apply.~~ LA03 M04

SR 3.2.2.1/SR 3.2.2.2 NOTE

d. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding

$F\Delta HR^M(X,Y) > BHNOM(X,Y)$

either of the following actions shall be taken:

SR 3.2.2.1/SR 3.2.2.2 NOTE a.

1. $F^M_{\Delta H}(X,Y)$ shall be increased over that specified in 4.2.3.2.a by the appropriate factor specified in the COLR, and 4.2.3.2.c.1 repeated, or

SR 3.2.2.1/SR 3.2.2.2 NOTE b.

2. $F^M_{\Delta H}(X,Y)$ shall be evaluated according to 4.2.3.2 at or before the time when the margin is projected to result in the action specified in 4.2.3.2.c.3 or 4.2.3.2.c.4.

SR 3.2.2.1 SR 3.2.2.2

4.2.3.3 $F^M_{\Delta H}(X,Y)$ shall be determined to be within its limit ~~by using the incore detectors to obtain a power distribution map~~: LA02

SR 3.2.2.1 SR 3.2.2.2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. ~~At least once per 31 EFPD.~~ ~~thereafter~~ In accordance with the Surveillance Frequency Control Program LA05 A04

~~* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA01

~~** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA03

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A01

ITS ITS 3.2.2

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 $F_{\Delta H}(X,Y)$ shall be maintained within the limits specified in the COLR.

Applicability APPLICABILITY: MODE 1

ACTION:

Add proposed ACTION A Note M01

ACTION A With $F_{\Delta H}(X,Y)$ exceeding the limit specified in the COLR:

a. Within 2 hours either:

1. ~~Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

Required Action A.1 2. Reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH~~*~~% for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and LA01

ACTION A b. Within the next ~~4~~ 72 hours either: L01

1. ~~Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

Required Action A.2 2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH~~*~~% for each 1% that $F_{\Delta H}(X,Y)$ exceeds that limit, and LA01

ACTION A c. Within 24 hours of initially being outside the limit specified in the COLR, either:

1. ~~Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or~~ A02

Required Action A.3 2. Verify ~~through incore flux mapping~~ that $F_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a.2 or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours. LA02

ACTION C L02

Add proposed ACTION C M02

~~* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA01

A01

ITS ITS 3.2.2

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

Required Action A.4

d. Within 48 hours of initially being outside the limit specified in the COLR, reduce the Overtemperature Delta T ~~K_1 term in Table 2.2-1~~ by at least TRH~~**~~ for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and LA03

Add proposed Required Action A.5 Note A03

Required Action A.5

e. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b. and/or c. and/or d., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}(X,Y)$ is demonstrated~~, through incore flux mapping~~, to be within the above limit prior to exceeding the following THERMAL POWER levels: LA02

Completion Time A.5

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

~~** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA03

A01

ITS ITS 3.2.2

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

SR NOTE

4.2.3.1 The provisions of Specification 4.0.4 are not applicable. M03

SR 3.2.2.1
SR 3.2.2.2

4.2.3.2 $F^{M}_{\Delta H}(X,Y)$ shall be evaluated to determine if $F_{\Delta H}(X,Y)$ is within its limit by:

~~a. Using the movable incore detectors to obtain a power distribution map~~ ~~$F^{M}_{\Delta H}(X,Y)$~~ ~~* at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Satisfying the following relationship:~~

~~$F\Delta HR^{M}(X,Y) \leq$ BHNOM (X,Y)~~

~~Where:~~

$$\cancel{F\Delta HR^{M}(X,Y) = \frac{F^{M}_{\Delta H}(X,Y)}{MAP^{M} / AXIAL(X,Y)}}$$

LA04

~~And BHNOM (X,Y)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.~~

~~MAP^{M} is the maximum Allowable Peak** obtained from the measured power distribution.~~

~~AXIAL (X,Y) is the axial shape for $F_{\Delta H}(X,Y)$.~~

c. If the above relationship is not satisfied, then

1. For the location, calculate the % margin to the maximum allowable design as follows:

SR 3.2.2.1

$$\% F_{\Delta H}\ Margin = \cancel{\left(1 - \frac{F\Delta HR^{M}(X,Y)}{BHDES(X,Y)}\right) x\ 100\%}$$

SR 3.2.2.2

$$\% f_{1}(\Delta I) Margin = \cancel{\left(1 - \frac{F\Delta HR^{M}(X,Y)}{BRDES(X,Y)}\right) x\ 100\%}$$

LA04

~~where BHDES(X,Y) and BRDES(X,Y)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within the LCO limits, and include allowances for calculational and measurement uncertainties.~~

* ~~No additional uncertainties are required in the following equations for $F^{M}_{\Delta H}(X,Y)$1 and $F\Delta HR^{M}(X,Y)$, because the limits include uncertainties.~~

LA04

** ~~BHNOM (X,Y), MAP^{M}, BHDES (X,Y), and BRDES (X,Y) data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.~~

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

~~2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above.~~

~~$F_{\Delta H}$ min margin = minimum % margin value of all locations examined~~

~~$f_1(\Delta I)$ min margin = minimum % margin value of all locations examined~~ LA04

ACTION A 3. If the $F_{\Delta H}$ min margin in 4.2.3.2.c.2 above is < 0, then within 2 hours reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH~~*~~% x most negative margin from 4.2.3.2.c.2 ~~and maintain the requirements of Specification 3.2.3; otherwise the Action statements for 3.2.3 apply.~~ LA01 M04

ACTION B 4. If the $f_1(\Delta I)$ min margin in 4.2.3.2.c.2 above is < 0, then within 48 hours reduce the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH~~**~~% x most negative margin from 4.2.3.2.c.2 ~~and maintain the requirements of Specification 3.2.3; otherwise the action statements for 3.2.3 apply.~~ LA03 M04

SR 3.2.2.1/SR 3.2.2.2 NOTE d. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding

$F\Delta HR^M (X,Y) > BHNOM (X,Y)$

either of the following actions shall be taken:

SR 3.2.2.1/SR 3.2.2.2 NOTE a. 1. $F^M_{\Delta H}(X,Y)$ shall be increased over that specified in 4.2.3.2.a by the appropriate factor specified in the COLR, and 4.2.3.2.c.1 repeated, or

SR 3.2.2.1/SR 3.2.2.2 NOTE b 2. $F^M_{\Delta H}(X,Y)$ shall be evaluated according to 4.2.3.2 at or before the time when the margin is projected to result in the action specified in 4.2.3.2.c.3 or 4.2.3.2.c.4.

SR 3.2.2.1 SR 3.2.2.2 4.2.3.3 $F^M_{\Delta H}(X,Y)$ shall be determined to be within its limit ~~by using the incore detectors to obtain a power distribution map~~: LA02

SR 3.2.2.1 SR 3.2.2.2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. ~~At least once per 31 EFPD.~~ In accordance with the Surveillance Frequency Control Program LA05 ~~thereafter~~ A04

~~* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA01

~~** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.~~ LA03

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.2.3 ACTION a.1, b.1 and c.1 require the restoration of $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR. ISTS LCO 3.0.2 Bases states that correction of the entered Condition is an action that may always be considered upon entering ACTIONS and that the restoration of compliance with the LCO is always an option. This changes the CTS by not specifically stating that restoration of $F_{\Delta H}(X,Y)$ is required.

This change is acceptable because the technical requirements have not changed. ISTS LCO 3.0.2 Bases states that correction of the entered Condition is an action that may always be considered upon entering ACTIONS and that the restoration of compliance with the LCO is always an available Required Action. The convention in the ITS is to not state such "restore" options explicitly unless it is the only action or is required for clarity. In this specific application, Required Action A.1.1 is not the only ACTION and a power reduction should be the focus for restoration of $F_{\Delta H}(X,Y)$ to within the limits. This change is designated as administrative, because it does not result in technical changes to the CTS.

A03 CTS 3.2.3 ACTION e states in part that with $F_{\Delta H}(X,Y)$ exceeding its limit, $F_{\Delta H}(X,Y)$ must be demonstrated to be within its limit prior to exceeding 50% RTP and 75% RTP, and within 24 hours of attaining or exceeding 95% RTP. ITS 3.2.2 Required Action A.5 contains the same requirements. However, ITS 3.2.2 Required Action A.5 is modified by a Note which states "THERMAL POWER does not have to be reduced to comply with this Required Action." This modifies the CTS by adding a Note stating that THERMAL POWER does not have to be reduced to comply with the Required Action.

This change is acceptable, because the requirements have not changed. The Note is included in the ITS to make clear that THERMAL POWER does not have to be reduced to perform the Required Action. For example, if $F_{\Delta H}(X,Y)$ exceeds its limit and, per ITS Required Action A.1, THERMAL POWER is reduced to 60% RTP, THERMAL POWER does not have to be reduced to less than 50% RTP to verify $F_{\Delta H}(X,Y)$ is within its limit to comply with ITS Required Action A.5. However, $F_{\Delta H}(X,Y)$ must still be measured prior to exceeding 75% RTP and within 24 hours of attaining or exceeding 95% RTP. The Note is needed because the ITS contains a Note in ITS 3.2.2 ACTION A that states "Required Actions A.3 and A.5 must be completed whenever Condition A is entered." The ITS 3.2.2 ACTION A Note does not exist in the CTS and could be construed as

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

requiring THERMAL POWER to be reduced to comply with Required Action A.5. (Addition of the ACTION A Note is discussed in DOC M01.) As a result, the Required Action A.5 Note makes the ITS and CTS actions consistent. This change is designated as administrative, because it does not result in technical changes to the CTS.

A04 CTS 4.2.3.3 requires $F^M_{\Delta H}(X,Y)$ to be determined prior to operation above 75% of RTP after each fuel loading, and at least once per 31 EFPD. ITS SR 3.2.2.1 and SR 3.2.2.2 Frequency is once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter. This changes the CTS by adding the word "thereafter" to the Frequency. The removal of the "31 EFPD thereafter" Frequency to the Surveillance Frequency Control Program is discussed in DOC LA05.

CTS 4.2.3.3 is required to be performed prior to operation above 75% RTP after each fuel loading and once per 31 EFPD. Also, although this Frequency is removed to the Surveillance Frequency Control Program, the addition of the word "thereafter" in ITS SR 3.2.2.1 and SR 3.2.2.2 ensures that the 31 EFPD Frequency starts after the first performance of the SR, which is required prior to exceeding 75% RTP after each fuel loading. Therefore, the addition of the word "thereafter" is considered acceptable because the use of "thereafter" is essentially the same as the CTS Frequency. This change is designated as administrative, because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.2.3 ACTION e states that with $F_{\Delta H}(X,Y)$ exceeding its limit "subsequent POWER OPERATION may proceed provided that $F_{\Delta H}(X,Y)$ is demonstrated, through incore flux mapping, to be within the above limit prior to exceeding the following THERMAL POWER levels: 1. A nominal 50% of RATED THERMAL POWER, 2. A nominal 75% of RATED THERMAL POWER, and 3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER." However, under CTS 3.0.2, these measurements do not have to be completed, if compliance with the LCO is restored. ITS 3.2.2 ACTION A contains a Note which states, "Required Actions A.3 and A.5 must be completed whenever Condition A is entered." ITS 3.2.2 Required Action A.3 requires verification that $F_{\Delta H}$ min margin is ≥ 0 24 hours after entry into Condition A. Required Action A.5 requires verification that $F_{\Delta H}$ min margin is ≥ 0 prior to THERMAL POWER exceeding 50% RTP and 75% RTP, and within 24 hours after THERMAL POWER is greater than or equal to 95% RTP. This changes the CTS by requiring the verification that $F_{\Delta H}$ min margin is ≥ 0 to be made even if $F_{\Delta H}(X,Y)$ is restored to within its limit. SII

This change is acceptable, because it establishes appropriate compensatory measurements for violation of the $F_{\Delta H}(X,Y)$ limit. As power is reduced under ITS 3.2.2 Required Action A.1, the margin to the $F_{\Delta H}(X,Y)$ limit increases. Therefore, compliance with the LCO could be restored during the power reduction. Verifying that the limit is met as power is increased ensures that the limit continues to be met and does not remain unmeasured for up to 31 EFPD. This change is

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

designated as a more restrictive change because it imposes requirements in addition to those in the CTS.

M02 CTS 3.2.3 does not contain an Action to follow if ACTIONS a, b, d, and e cannot be met. Therefore, CTS 3.0.3 would be entered, which would allow 1 hour to initiate a shutdown and to be in HOT STANDBY within 7 hours. ITS 3.2.2 ACTION C, states that the plant must be in MODE 2 within 6 hours, if any Required Action and associated Completion Time is not met. This changes the CTS by eliminating the one hour to initiate a shut down and, consequently, allowing one hour less for the unit to be in MODE 2.

The purpose of CTS 3.0.3 is to delineate the ACTION to be taken for circumstances not directly provided for in the ACTION statement and whose occurrences would violate the intent of the Specification. This change is acceptable because it provides an appropriate compensatory measure for the described conditions. If any Required Action and associated Completion Time cannot be met, the unit must be placed in a MODE in which the LCO does not apply. The LCO is applicable in MODE 1. Requiring a shut down to MODE 2 is appropriate in this condition. The one hour allowed by CTS 3.0.3 to prepare for a shut down is not needed, because the operators have had time to prepare for the shut down while attempting to follow the Required Actions and associated Completion Times. This change is designated as more restrictive because it allows less time to shut down than does the CTS.

M03 CTS 4.2.3.1 ″The provisions of Specification 4.0.4 are not applicable″ provides an allowance for entering the next higher MODE of Applicability when the LCO is not met. ITS 3.2.2 has no specific allowance for changing MODES at any time with ITS LCO 3.2.2 not met. ITS LCO 3.0.4 requires in part that, ″When an LCO is not met, entry into a MODE or other specified Condition in the Applicability shall only be made: If either part a. or part b. or part c. is met.″ Part c provides the following allowance, ″When an allowance is stated in the individual value, parameter or other specification.″ ITS 3.2.2 Surveillance Requirements Note will be added to provide the following allowance, ″Not required to be performed until 12 hours after an equilibrium power level has been achieved, at which a power distribution map can be obtained.″ This changes CTS by allowing entry into the MODE of Applicability by only deferring the performance of the Surveillance Requirements instead of deferring compliance with the LCO.

The purpose of CTS 4.2.3.1 is to provide an exception to SR 4.0.4. SR 4.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. This change is acceptable because ITS provides an allowance to enter the MODE of Applicability at any time ITS LCO 3.2.2 is not met solely based on surveillance performance. SR 3.2.2.1 and SR 3.2.2.2 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level (~40%-50%). The surveillances cannot be performed in MODE 2. This change is designated as more restrictive because the CTS 4.0.4 MODE change allowance for ″not met″ is now limited to

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}$(X,Y)

the performance of the SRs and does not include the allowance to change MODES with the acceptance criteria not met.

M04 CTS 3.2.3 provides two acceptable alternatives for the $F_{\Delta H}$ min margin and $f_1(\Delta I)$ min margin not met. CTS 4.2.3.2.c.3 states, ″If the $F_{\Delta H}$ min margin in 4.2.3.2.c.2 above is < 0, then within 2 hours reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH*% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the Action statements for 3.2.3 apply.″ CTS 4.2.3.2.c.4 states, ″ If the $f_1(\Delta I)$ min margin in 4.2.3.2.c.2 above is < 0, then within 48 hours reduce the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH**% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the action statements for 3.2.3 apply.″ CTS 4.2.3.2.c.3 has been replaced by ITS 3.2.2 Required Actions A.1. Similarly, CTS 4.2.3.2.c.4 has been replaced with ITS 3.2.2 Required Actions B.1. However, in both cases the option for, ″ otherwise, the action statements for 3.2.3 apply ″ has not been retained. This changes the CTS by removing the option to follow the action statement of CTS 3.2.3 for either min margin ($F_{\Delta H}$ or $f_1(\Delta I)$) not met.

The purpose of CTS 4.2.3.2.c.3 and CTS 4.2.3.2.c.4 is to provide acceptable alternatives for the required compensatory actions when either $F_{\Delta H}$ min margin or $f_1(\Delta I)$ min margin is not met. The CTS surveillance requirements for $F_{\Delta H}$ min margin not met requires the reduction of ALLOWABLE THERMAL POWER from RTP by RRH*% x most negative margin from 4.2.3.2.c.2. This requirement is being retained as ITS 3.2.2 Required Action A.1. The CTS surveillance requirements for $f_1(\Delta I)$ min margin not met requires the reduction of the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH**% x most negative margin from 4.2.3.2.c.2. This requirement is being retained as ITS 3.2.2 Required Action B.1. If the ITS Required Actions are not performed within the allowed Completion Time, Condition C will be entered requiring the Unit to be placed in MODE 2. This change is designated as more restrictive because an acceptable alternative Required Action available in CTS is being removed.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.3 provides actions to take within 2 hours when $F_{\Delta H}$(X,Y) is not within limits, and states to reduce the allowable THERMAL POWER and within 4 hours reduce the Power Range Neutron Flux-High Trip Setpoint at least RRH*% for each 1% that $F_{\Delta H}$(X,Y) exceeds the limit provided in the COLR. Similarly, CTS 4.2.3.2.c.3 requires in part to reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH*% x most negative margin from 4.2.3.2.c.2. CTS NOTE * provides the definition of RRH as the amount of power reduction required to compensate for each 1% that $F_{\Delta H}$(X,Y)

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

exceeds the limit provided in the COLR per Specification 6.9.1.14. ITS 3.2.2 Required Action A.1 requires within 2 hours of discovery that $F_{\Delta H}$ min margin is not within limits, to reduce THERMAL POWER from RTP, and ITS 3.2.2 Required Action A.2 requires within 72 hours to reduce the Power Range Neutron Flux-High Trip Setpoint by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin. This changes the CTS by relocating the definition of RRH to the COLR.

The removal of these details from the Technical Specifications and its relocation into the COLR is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements to reduce THERMAL POWER from RTP and reduce the Power Range Neutron Flux-High Trip Setpoint by ≥ RRH% for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit. The definition of RRH is already located in the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the COLR requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.3 ACTIONS c.2 and e require $F_{\Delta H}(X,Y)$ to be determined to be within its limit through incore flux mapping. Additionally, CTS 4.2.3.3 requires $F^{M}_{\Delta H}(X,Y)$ to be determined to be within its limit by using the incore detectors to obtain a power distribution map. ITS SR 3.2.2.1 and SR 3.2.2.2 collectively verifiy that $F_{\Delta H}(X,Y)$ is within its limit. This changes the CTS by moving the manner in which the $F_{\Delta H}(X,Y)$ determination is performed to the Bases.

The removal of these details for performing actions and a Surveillance Requirement from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to determine $F_{\Delta H}(X,Y)$ is within its limit. Also, this change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.3 Action d requires within 48 hours of $F_{\Delta H}(X,Y)$ being outside its limits, to reduce the Overtemperature Delta T K_1 term in Table 2.2-1 by at least TRH** for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit. Similarly, CTS 4.2.3.2.c.4 requires in part to reduce Overtemperature Delta T K1 term in

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

Table 2.2-1 by at least TRH** x most negative margin from 4.2.3.2.c.2. CTS Note ** provides a definition for TRH as the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR. ITS 3.2.2 Required Action A.4 states when $F_{\Delta H}$ min margin is < 0, reduce the OTΔT setpoint by ≥ TRH multiplied times the $f_1(\Delta I)$ min margin. This changes the CTS by moving the details of the specific variable within OTΔT to be reduced, the location of the K_1 terms, and the definition of TRH to the COLR.

The removal of these details from the Technical Specifications and their relocation into the COLR is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to reduce the OTΔT setpoint by ≥ TRH multiplied times the $f_1(\Delta I)$ min margin. The specific variable within OTΔT to be reduced, the location of the K_1 terms, and definition of TRH are already located in the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the COLR requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.2.3.2.a, 4.2.3.2.b, 4.2.3.2.c.1, and 4.2.3.2.c.2, provide details for evaluating $F^M_{\Delta H}(X,Y)$ to determine if $F_{\Delta H}(X,Y)$ is within limits. ITS SR 3.2.2.1 and SR 3.2.2.2 collectively verify that $F_{\Delta H}(X,Y)$ is within limits specified in the COLR. This changes the CTS by moving the details for evaluating $F^M_{\Delta H}(X,Y)$ to determine if $F_{\Delta H}(X,Y)$ is within limits to the TS Bases.

The removal of these details from the Technical Specifications and their relocation into the TS Bases is acceptable, because the procedural steps and further details for making a determination that $F_{\Delta H}(X,Y)$ is within its limits is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to determine $F_{\Delta H}(X,Y)$ is within its limits specified in the COLR. Also, this change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA05 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.2.3.3 requires, in part, a determination that $F_{\Delta H}(X,Y)$ is within its limits at least once per 31 EFPD. ITS SR 3.2.2.1 and SR 3.2.2.2 collectively require a similar Surveillance and specify the periodic Frequency as, "In

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (Category 3 – Relaxation of Completion Time) CTS 3.2.3 ACTION b states, in part, that when $F_{\Delta H}(X,Y)$ exceeds its limit, reduce the Power Range Neutron Flux - High Trip setpoints by at least RRH*% for each 1% $F_{\Delta H}(X,Y)$ exceeds that limit within the next 4 hours. ITS 3.2.2 Required Actions A.2 states with $F_{\Delta H}(X,Y)$ not within limit, reduce the Power Range Neutron Flux - High trip setpoints by at least RRH% multiplied times the $F_{\Delta H}$ min margin within 72 hours. This changes the CTS by increasing the time allowed to reduce the trip setpoints.

The purpose of CTS 3.2.3 ACTION b is to lower the Power Range Neutron Flux - High Trip setpoints, which ensures continued operation is at an acceptably low power level with an adequate DNBR margin and avoids violating the $F_{\Delta H}(X,Y)$ limit. This change is acceptable, because the Completion Time is consistent with safe operation and recognizes that the safety analysis assumptions are satisfied once power is reduced, and considers the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Time allows the Power Range Neutron Flux - High Trip setpoints to be reduced in a controlled manner without challenging operators, technicians, or plant systems. Following a significant power reduction, a time period of 24 hours is allowed to reestablish steady state xenon concentration and power distribution and to take and analyze a flux map. If it is determined that $F_{\Delta H}(X,Y)$ is still not within its limit, reducing the Power Range Neutron Flux - High Trip Setpoints can be accomplished within a few hours. Furthermore, setpoint changes should only be required for extended operation in this condition, because of the risk of a plant trip during the adjustment. This change is designated as less restrictive, because additional time is allowed to lower the Power Range Neutron Flux - High Trip setpoints than was allowed in the CTS.

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

L02 (Category 3 – Relaxation of Completion Time) CTS 3.2.3 ACTION c.2 states, "Verify through incore flux mapping that $F_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a.2 or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next two hours." ITS 3.2.2 ACTION C states, "Required Action and associated Completion Time not met." Required Action C.1 states, "Be in MODE 2" within a Completion Time of "6 hours." This changes the CTS by increasing the time allowed to exit the MODE of Applicability when the Required Actions or associated Completion Times are not met.

The purpose of CTS 3.2.3 ACTION c.2 is to, within 24 hours, either verify $F_{\Delta H}(X,Y)$ is restored within limits for the reduced power level or within the next 2 hours, enter MODE 2. Under similar conditions, ITS will require the plant to be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems. This change is acceptable, because the Completion Time is consistent with safe operation and recognizes that the safety analysis assumptions are satisfied once power is reduced. This change is designated as less restrictive, because additional time is allowed to exit the LCO than was allowed in the CTS.

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

CTS

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
3.2.2

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~) 1

3.2.3 LCO 3.2.2 $F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ shall be within the limits specified in the COLR. 1

Applicability APPLICABILITY: MODE 1.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
DOC M01 ACTION a.2 SR 4.2.3.2.c.3 SR 4.2.3.2.c.3	A. ------------NOTE------------ Required Actions A.~~2~~ 3 and A.~~3~~ 5 must be completed whenever Condition A is entered. --------------------------------- ~~$F^N_{\Delta H}$ not within limit.~~ $F_{\Delta H}$ min margin < 0	~~A.1.1 Restore $F^N_{\Delta H}$ to within limit.~~ ~~OR~~ A.1.~~2.1~~ Reduce allowable THERMAL POWER ~~to < 50% RTP~~ INSERT 1. AND	~~4 hours~~ ~~4~~ 2 hours	2 3 2 4 5
ACTION b.2		A.~~1.2.~~2 Reduce Power Range Neutron Flux - High trip setpoints ~~to ≤ 55% RTP~~ INSERT 2.	72 hours	6 2 5
ACTION c.2		AND A.~~2~~ 3 Perform SR 3.2.2.1	24 hours	2
		AND INSERT 3		3

5 **INSERT 1**

ACTION a.2
4.2.3.2.c.3

from RTP by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin.

5 **INSERT 2**

ACTION b.2

by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin.

3 **INSERT 3**

ACTION d

A.4	Reduce Overtemperature ΔT trip setpoint by ≥ TRH multiplied times the $F_{\Delta H}$ min margin.	48 hours
AND		

CTS

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~
3.2.2

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION e DOC A03	A.~~3~~ 5 --------------NOTE-------------- THERMAL POWER does not have to be reduced to comply with this Required Action. -------------------------------------	3
ACTION e.1	Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND
ACTION e.2		Prior to THERMAL POWER exceeding 75% RTP AND
ACTION e.3 INSERT 4		24 hours after THERMAL POWER reaching ≥ 95% RTP 6
ACTION c.2 DOC M02 C ~~B~~. Required Action and associated Completion Time not met.	C ~~B~~.1 Be in MODE 2.	6 hours 6

6

INSERT 4

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. $f_1(\Delta I)$ min margin < 0.	B.1	Reduce Overtemperature ΔT trip setpoint by ≥ TRH multiplied times the $f_1(\Delta I)$ min margin.	48 hours

4.2.3.2.c.4

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CTS

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
3.2.2

SURVEILLANCE REQUIREMENTS INSERT 5 7

	SURVEILLANCE	FREQUENCY
4.2.3.2.c.1 4.2.3.3.a 4.2.3.3.b	SR 3.2.2.1 Verify ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin ~~≥~~ $\geq$ 0 ~~is within limits specified in the COLR.~~ INSERT 6 8	Once after each refueling prior to THERMAL POWER exceeding 75% RTP 8 SII AND ~~[31 EFPD thereafter~~ 9 ~~OR~~ 9 In accordance with the Surveillance Frequency Control Program ~~]~~

INSERT 7 8

7 **INSERT 5**

---NOTE---

4.2.3.1 Not required to be performed until 12 hours after an equilibrium power level has been achieved, at which a power distribution map can be obtained.

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CTS 3.2.2

INSERT 6

-------------------------------NOTE------------------------------ 4.2.3.2.d If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F\Delta HR^M(X,Y) > BHNOM(X,Y)$ 4.2.3.2.d.1 a. Increase $F^M_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F_{\Delta H}$ min margin ~~>~~ ≥ 0; or 4.2.3.2.d.2 b. Repeat SR 3.2.2.1 prior to the time at which the projected $F_{\Delta H}$ min margin will be < 0. ---	SII

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8 **INSERT 7**

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	------------------------------NOTE------------------------------ 4.2.3.2.d If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F\Delta HR^{M}(X,Y) > BHNOM(X,Y)$ 4.2.3.2.d.1 a. Increase $F^{M}_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $f_1(\Delta I)$ min margin ~~>~~ 0; or ≥ SII 4.2.3.2.d.2 b. Repeat SR 3.2.2.2 prior to the time at which the projected $f_1(\Delta I)$ min margin will be < 0. -- 4.2.3.2.c.1 Verify $f_1(\Delta I)$ min margin ~~>~~ 0. ≥ SII	Once after each refueling prior to THERMAL 4.2.3.3.a POWER exceeding 75% RTP 4.2.3.3.b <u>AND</u> In accordance with the Surveillance Frequency Control Program

CTS

$F_{\Delta H}$(X,Y) ~~$F^N_{\Delta H}$~~ 1
3.2.2

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$(X,Y) ~~$F^N_{\Delta H}$~~) 1

3.2.3 LCO 3.2.2 $F_{\Delta H}$(X,Y) ~~$F^N_{\Delta H}$~~ shall be within the limits specified in the COLR. 1

Applicability APPLICABILITY: MODE 1.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
DOC M01	A. ------------NOTE------------ Required Actions A.~~2~~ 3 and A.~~3~~ 5 must be completed whenever Condition A is entered. --------------------------------	~~A.1.1 Restore $F^N_{\Delta H}$ to within limit.~~	~~4 hours~~	2 3
		~~OR~~		
ACTION a.2 SR 4.2.3.2.c.3		A.1.~~2.1~~ Reduce allowable THERMAL POWER ~~to < 50% RTP~~ INSERT 1.	2 ~~4~~ hours	2 4 5
SR 4.2.3.2.c.3	~~$F^N_{\Delta H}$ not within limit.~~ $F_{\Delta H}$ min margin < 0	AND		
ACTION b.2		A.~~1.2.~~2 Reduce Power Range Neutron Flux - High trip setpoints ~~to ≤ 55% RTP~~ INSERT 2.	72 hours	2 6 5
		AND		
ACTION c.2		A.~~2~~ 3 Perform SR 3.2.2.1	24 hours	2
		AND INSERT 3		3

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CTS 3.2.2

5 **INSERT 1**

ACTION a.2
4.2.3.2.c.3

from RTP by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin.

5 **INSERT 2**

ACTION b.2

by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin.

3 **INSERT 3**

ACTION d

A.4	Reduce Overtemperature ΔT trip setpoint by ≥ TRH multiplied times the $F_{\Delta H}$ min margin.	48 hours
AND		

CTS

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
3.2.2

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION e DOC A03		A.~~3~~ 5	--------------NOTE-------------- THERMAL POWER does not have to be reduced to comply with this Required Action. -------------------------------------		3
ACTION e.1			Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND	
ACTION e.2				Prior to THERMAL POWER exceeding 75% RTP AND	
ACTION e.3	INSERT 4			24 hours after THERMAL POWER reaching ≥ 95% RTP	6
ACTION c.2 DOC M02	C ~~B~~. Required Action and associated Completion Time not met.	C ~~B~~.1	Be in MODE 2.	6 hours	6

6 **INSERT 4**

CONDITION	REQUIRED ACTION	COMPLETION TIME
4.2.3.2.c.4 B. $f_1(\Delta I)$ min margin < 0.	B.1 Reduce Overtemperature ΔT trip setpoint by ≥ TRH multiplied times the $f_1(\Delta I)$ min margin.	48 hours

CTS

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
3.2.2

SURVEILLANCE REQUIREMENTS INSERT 5 7

SURVEILLANCE		FREQUENCY
4.2.3.2.c.1 4.2.3.3.a 4.2.3.3.b SR 3.2.2.1 SII	INSERT 6 8 Verify ~~$F^N_{\Delta H}$~~ ~~is within limits specified in the COLR.~~ $F_{\Delta H}$ min margin $\geq$ 0 $\geq$	Once after each refueling prior to THERMAL POWER exceeding 75% RTP 8 AND ~~[31 EFPD thereafter~~ 9 ~~OR~~ 9 In accordance with the Surveillance Frequency Control Program ~~]~~

INSERT 7 8

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7 **INSERT 5**

---NOTE---

4.2.3.1 Not required to be performed until 12 hours after an equilibrium power level has been achieved, at which a power distribution map can be obtained.

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CTS 3.2.2

INSERT 6

-----------------------------NOTE----------------------------- 4.2.3.2.d If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F\Delta HR^M(X,Y) > BHNOM(X,Y)$ 4.2.3.2.d.1 a. Increase $F_{\Delta H}^{M}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F_{\Delta H}$ min margin ~~>~~ $\boxed{\geq}$ 0; or 4.2.3.2.d.2 b. Repeat SR 3.2.2.1 prior to the time at which the projected $F_{\Delta H}$ min margin will be < 0. --	

SII

Insert Page 3.2.2-3b

CTS

3.2.2

8 **INSERT 7**

CTS	SURVEILLANCE		FREQUENCY	
4.2.3.2.d	SR 3.2.2.2	-------------------------------NOTE------------------------------ If two measurements extrapolated to 31 EFPD beyond the most recent measurement yield: $F\Delta HR^{M}(X,Y) > BHNOM(X,Y)$		
4.2.3.2.d.1		a. Increase $F^{M}_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify f_1 (ΔI) min margin > 0; or ≥		SII
4.2.3.2.d.2		b. Repeat SR 3.2.2.2 prior to the time at which the projected f_1 (ΔI) min margin will be < 0. ---		
4.2.3.2.c.1 4.2.3.3.a 4.2.3.3.b		Verify f_1(ΔI) min margin > 0. ≥	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND In accordance with the Surveillance Frequency Control Program	SII

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}$(X,Y))

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS LCO 3.2.2 Required Action A.1.1 states, restore $F^N_{\Delta H}$ to within limit. ITS 3.2.2 will not retain the specific requirement to restore. LCO 3.0.2 Bases states that correction of the entered Condition is an action that may always be considered upon entering ACTIONS. This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action. The convention in the ITS is to not state such "restore" options explicitly unless it is the only action or is required for clarity. In this specific application, Required Action A.1.1 is not the only ACTION and a power reduction should be the focus for restoration of $F_{\Delta H}$(X,Y)to within the limits. Subsequent Required Actions have been renumbered o reflect this deletion.

3. Required Action A.4 is added to the ITS. CTS 3.2.3 ACTION d requires reduction of the OTΔT setpoint when $F_{\Delta H}$(X,Y) exceeds the limit in the COLR. Subsequent Required Actions have been renumbered o reflect this deletion.

4. The Completion Times for reducing THERMAL POWER upon discovery that $F_{\Delta H}$(X,Y) has exceeded its limit are shortened from 4 hours to 2 hours consistent with the current licensing basis.

5. The amount that THERMAL POWER and the Power Range Neutron Flux – High Trip setpoints are reduced after $F_{\Delta H}$(X,Y) has exceeded its limit are changed to reflect the values in the current licensing basis.

6. ITS Conditions A and B have been changed to reflect the CTS ACTIONs for both $F_{\Delta H}$ and/or $f_1(\Delta I)$ min margins not met.

7. ISTS LCO 3.2.2 does not contain a specific provision for changing MODES if LCO 3.2.2 is not met, other than the generic use of LCO 3.0.4. CTS SR 4.2.3.1 states, "The provisions of Specification 4.0.4 are not applicable." This allowance enables SQN to enter the MODE of Applicability with the Surveillance not met or performed. SQN is retaining the allowance to change the MODE of Applicability with the Surveillance not performed by adding a Surveillance Note to retain the allowance.

SII

8. ISTS SR 3.2.2.1 and SR 3.2.2.2 have been changed to reflect the CTS evaluation of $F_{\Delta H}$ min margin $\geq$ 0 and $f_1(\Delta I)$ min margin $\geq$ 0.

9. ISTS SR 3.2.2.1 (and proposed ITS SR 3.2.2.2) provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$) 1

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, ~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is a measure of the maximum total power produced in a fuel rod. 1

~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. ~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ typically increases with control bank insertion and typically decreases with fuel burnup. 1

~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine ~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ INSERT 1. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. 2

The COLR provides peaking factor limits that ensure that the design basis value of the ~~departure from nucleate boiling (~~DNB~~)~~ is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to ~~[1.3] using the [W3] CHF~~ the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an ~~$F^{N}_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ value that satisfies the LCO requirements. 1 1

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

2 **INSERT 1**

An $F_{\Delta H}(X,Y)$ evaluation requires obtaining an incore flux map in MODE 1. The incore flux map results provide the measured value ($F^{M}_{\Delta H}(X,Y)$ of $F_{\Delta H}(X,Y)$ for each assembly location (X,Y). The $F_{\Delta H}$ ratio (FDHR) is used in order to determine the $F_{\Delta H}$ limit for the measured and design power distributions (Ref. 4). Then,

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$$F\Delta HR^{M}(X,Y) = \frac{F^{M}_{\Delta H}(X,Y)}{MAP^{M} / AXIAL^{M}(X,Y)}$$

where MAP^{M} is the maximum allowable peak from the COLR for the measured assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties, and $AXIAL^{M}(X,Y)$ is the measured ratio of the peak-to-average axial power at assembly location (X,Y).

BHDES(X,Y) is a cycle dependent design limit to preserve Departure from Nucleate Boiling(DNB) assumed for initial conditions at the time of limiting transients such as a Loss of Flow Accident (LOFA). BRDES(X,Y) is a cycle dependent design limit to preserve reactor protection system safety limits for DNB requirements (Ref. 4).

The expression for BHDES(X,Y) is:

$$BHDES(X,Y) = F\Delta HR^{d}(X,Y) * MH(X,Y)$$

where: $F\Delta HR^{d}(X,Y) = \dfrac{F^{d}_{\Delta H}(X,Y)}{MAP^{d} / AXIAL^{d}(X,Y)}$

- MAP^{d} is the maximum allowable peak from the COLR for the design assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties,

- $AXIAL^{d}(X,Y)$ is the design ratio of the peak-to-average axial power at assembly location (X,Y),

- $F^{d}_{\Delta H}(X,Y)$ is the design $F_{\Delta H}$ assembly location (X, Y), and

- MH(X,Y) is the minimum available margin ratio for initial condition DNB at the limiting conditions at assembly location (X,Y).

2 **INSERT 1 (continued)**

The expression for BRDES(X,Y) is:

$BRDES(X,Y) = F\Delta HR^{d}(X,Y) * MH^{s}(X,Y)$

where: $MH^{s}(X,Y)$ is the minimum available margin ratio for steady state DNB at the limiting conditions at assembly location (X,Y).

The reactor core is ″operating as designed″ if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is ″operating as designed″ if the following relationship is satisfied:

$F\Delta HR^{M}(X,Y) \leq BHNOM(X,Y)$

where: BHNOM(X,Y) is the nominal design radial peaking factor for an assembly at core location (X,Y) increased by an allowance for the expected deviation between the measured and predicted design power distribution.

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1

B 3.2.2

BASES

APPLICABLE SAFETY ANALYSES

Limits on ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits: 1

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm ~~[~~(Ref. 1~~]~~), and 3 8

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum ~~DNBR to the 95/95 DNB criterion of [1.3] using the [W3] CHF~~ local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB. 1

The allowable ~~$F^N_{\Delta H}$ limit~~ $F_{\Delta H}(X,Y)$ limits, $F_{\Delta H}$ min margin and $f_1(\Delta I)$ min margin, increase~~s~~ with decreasing power level. This functionality in ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ as a function of power level defined by the COLR limit equation. 1

The LOCA safety analysis indirectly models ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F_Q(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature ~~[~~(Ref. 3~~]~~). 3 8

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

(QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ~~($F^N_{\Delta H}$)~~ $F_{\Delta H}(X,Y)$," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F_Q(~~Z~~ X, Y, Z))."

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ and F_Q(~~Z~~ X, Y, Z) are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

LCO

~~$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.~~

INSERT 2

~~The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.~~

~~The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.~~

~~A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of is $F^N_{\Delta H}$ allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.~~

2

1

APPLICABILITY

The ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in these modes.

1

2 **INSERT 2**

SII

The LCO states that $F_{\Delta H}(X,Y)$ shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured $F_{\Delta H}(X,Y)$ to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing $F\Delta HR^M(X,Y)$ to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the $F_{\Delta H}$ min margin is ≥ 0 and $f_1(\Delta I)$ min margin ≥ 0 the LCO is satisfied.

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$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
B 3.2.2

BASES INSERT 3 2

ACTIONS ~~A.1.1~~

~~With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.~~ 4

Condition A is modified by a Note that requires that Required Actions A.~~2~~ 3 and A.~~3~~ 5 must be completed whenever Condition A is entered. ~~Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.~~ INSERT 4 4

However, if power is reduced below 50% RTP, Required Action A.~~3~~ 5 requires that another determination of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin must be ~~done~~ verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. ~~In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.~~ 4 1 5

A.1~~.2.1~~ and A.~~1.2.~~2

If the value of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce INSERT 5 THERMAL POWER ~~to < 50% RTP~~ in accordance with Required Action A.1~~.2.1~~ and reduce the Power Range Neutron Flux - High INSERT 6 ~~to ≤ 55% RTP~~ in accordance with Required Action A.~~1.2.~~2. Reducing INSERT 7 allowable RTP ~~to < 50% RTP~~ increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of ~~4~~ 2 hours for Required Action A.1~~.2.1 is consistent with those allowed for in Required Action A.1.1 and~~ provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. ~~The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive~~. 1 4 3 3

2 **INSERT 3**

The % $F_{\Delta H}$ margin is based on the relationship between $F\Delta HR^M(X,Y)$ and the limit, BHDES (X,Y), as follows:

$$\% \ F_{\Delta H} \text{ Margin } = \left(1 - \frac{F\Delta HR^{M}(X, Y)}{BHDES(X, Y)}\right) x \ 100\%$$

If the reactor core is ″operating as designed″, then $F\Delta HR^M(X,Y)$ is less than BHDES (X,Y) and calculation of %$F_{\Delta H}$ margin is not required. If the %$F_{\Delta H}$ margin is less than zero, then $F\Delta HR^M(X,Y)$ is greater than BHDES (X, Y) and the $F_{\Delta H}(X,Y)$ limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial conditions DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The $F_{\Delta H}$ min margin is the minimum for all core locations examined.

4 **INSERT 4**

If $F_{\Delta H}$ min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance of Required Actions A.3 and A.5 must be met.

4 **INSERT 5**

from RTP by at least RRH % (where RRH = Thermal power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit) multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 6**

trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 7**

by at least RRH% multiplied times the $F_{\Delta H}$ min margin

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1

B 3.2.2

BASES

ACTIONS (continued)

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.~~1.2.~~2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System. 4

A.~~2~~ 3

allowable INSERT 8

SII $F_{\Delta H}$ min margin is verified $\geq$ 0

Once the power level has been reduced ~~to < 50% RTP~~ per Required Action A.1~~.2.1~~, an incore flux map (SR 3.2.2.1) must be obtained and the ~~measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit~~ at the lower power level. The unit is provided ~~2~~0 additional hours to perform this task over and above the 4 hours allowed by ~~either Action A.1.1 or~~ Action A.1~~.2.1~~. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate ~~$F^N_{\Delta H}$~~. 4 1 3 22 2

$F_{\Delta H}$ min margin 1

INSERT 9 4

A.~~3~~ 5 4

$\geq$ $F_{\Delta H}$ min margin is ~~>~~ 0

Verification that ~~$F^N_{\Delta H}$ is within its specified limits~~ after an out of limit occurrence ensures that the cause that led to the ~~$F^N_{\Delta H}$~~ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the ~~$F^N_{\Delta H}$~~ limit is ~~within the LCO limits~~ prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. 1

$F_{\Delta H}$ min margin $F_{\Delta H}$ min margin ~~>~~0 $\geq$

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

INSERT 10 2

C ~~B~~.1 4

5, and B.1,

When Required Actions A.1~~.1~~ through A.~~3~~ cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

4 **INSERT 8**

by at least RRH% multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 9**

A.4

If the value of $F\Delta HR^M(X,Y)$ is not restored to within its specified limit, Overtemperarture ΔT K1 (OTΔT K1) term is required to be reduced by at least TRH multiplied times the $F_{\Delta H}$ min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

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2 **INSERT 10**

B.1

The $\%f_1(\Delta I)$ margin is based on the relationship between $F\Delta HR^M(X,Y)$ and the limit, BRDES (X,Y), as follows:

$$\% \ f_1(\Delta I)Margin = \left(1 - \frac{F\Delta\ HR^M(X,Y)}{BRDES(X,Y)}\right) x\ 100\%$$

If the reactor core is ″operating as designed″, then $F\Delta HR^M(X,Y)$ is less than BRDES (X,Y) and calculation of $\%f_1(\Delta I)$ margin is not required. If the $\%f_1(\Delta I)$ margin is less than zero, then $F\Delta HR^M(X,Y)$ is greater than BRDES (X, Y) and the OTΔT setpoint limits may not be adequate to prevent exceeding DNB requirements.

BRDES (X,Y) represents the maximum allowable design radial peaking factors which ensure that the steady state DNBR limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties

Required Action B.1 requires the reduction of the OTΔT K1 term by at least TRH multiplied by the $f_1(\Delta I)$ min margin. TRH is the amount of OTΔT K1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR. Completing Required Action B.1 within the allowed Completion Time of 48 hours, restricts $F_{\Delta H}(X,Y)$ such that even if a transient occurred, DNB requirements are met. The $f_1(\Delta I)$ min margin is the minimum % of $f_1(\Delta I)$ margin for all core locations examined.

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1

B 3.2.2

BASES

SURVEILLANCE REQUIREMENTS

INSERT 11 4

~~SR 3.2.2.1~~

~~The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.~~

~~After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.~~ 1

~~[The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.~~ 6

~~OR~~

INSERT 12 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7

~~---]~~

REFERENCES

1. Regulatory Guide 1.77, Rev. ~~[~~0~~]~~, May 1974.

2. 10 CFR 50, Appendix A, GDC 26.

3. 10 CFR 50.46.

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4. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.

4 **INSERT 11**

SR 3.2.2.1 and SR 3.2.2.2 are modified by a Note. It states that, "Not required to be performed until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained." SR 3.2.2.1 and SR 3.2.2.2 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level. These surveillances could not be satisfactorily performed if the requirement for performance of the Surveillances was included in MODE 2 prior to entering MODE 1.

In a reload core, $F^{M}_{\Delta H}(X,Y)$ could not have previously been measured, therefore, there is a Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F^{M}_{\Delta H}(X,Y)$ is made at a lower power level at which adequate margin is available before going to 100% RTP.

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4 **INSERT 12**

SR 3.2.2.1 and SR 3.2.2.2

In addition to ensuring via Surveillance that the nuclear enthalpy rise hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in $F^{M}_{\Delta H}(X,Y)$ for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an $F\Delta HR^{M}(X,Y) > BHNOM(X,Y)$, further consideration is required.

The implications of these extrapolations are considered separately for BHDES(X,Y) and BRDES(X,Y) limits. If the extrapolations of $F^{M}_{\Delta H}(X,Y)$ are unfavorable, additional actions must be taken. These actions are to meet the $F_{\Delta H}(X,Y)$ limit with the last $F^{M}_{\Delta H}(X,Y)$ increased by the appropriate factor specified in the COLR or to evaluate $F^{M}_{\Delta H}(X,Y)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$) 1

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is a measure of the maximum total power produced in a fuel rod. 1

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ typically increases with control bank insertion and typically decreases with fuel burnup. 1

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ INSERT 1. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. 2

The COLR provides peaking factor limits that ensure that the design basis value of the ~~departure from nucleate boiling (~~DNB~~)~~ is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to ~~[1.3] using the [W3] CHF~~ the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ value that satisfies the LCO requirements. 1 1

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

2

INSERT 1

An $F_{\Delta H}(X,Y)$ evaluation requires obtaining an incore flux map in MODE 1. The incore flux map results provide the measured value ($F^{M}_{\Delta H}(X,Y)$ of $F_{\Delta H}(X,Y)$ for each assembly location (X,Y). The $F_{\Delta H}$ ratio (FDHR) is used in order to determine the $F_{\Delta H}$ limit for the measured and design power distributions (Ref. 4). Then,

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$$F\Delta HR^{M}(X,Y) = \frac{F^{M}_{\Delta H}(X,Y)}{MAP^{M} / AXIAL^{M}(X,Y)}$$

where MAP^{M} is the maximum allowable peak from the COLR for the measured assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties, and $AXIAL^{M}(X,Y)$ is the measured ratio of the peak-to-average axial power at assembly location (X,Y).

BHDES(X,Y) is a cycle dependent design limit to preserve Departure from Nucleate Boiling(DNB) assumed for initial conditions at the time of limiting transients such as a Loss of Flow Accident (LOFA). BRDES(X,Y) is a cycle dependent design limit to preserve reactor protection system safety limits for DNB requirements (Ref. 4).

The expression for BHDES(X,Y) is:

$$BHDES(X,Y) = F\Delta HR^{d}(X,Y) * MH(X,Y)$$

where: $F\Delta HR^{d}(X,Y) = \dfrac{F^{d}_{\Delta H}(X,Y)}{MAP^{d} / AXIAL^{d}(X,Y)}$

- MAP^{d} is the maximum allowable peak from the COLR for the design assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties,

- $AXIAL^{d}(X,Y)$ is the design ratio of the peak-to-average axial power at assembly location (X,Y),

- $F^{d}_{\Delta H}(X,Y)$ is the design $F_{\Delta H}$ assembly location (X, Y), and

- MH(X,Y) is the minimum available margin ratio for initial condition DNB at the limiting conditions at assembly location (X,Y).

2 **INSERT 1 (continued)**

The expression for BRDES(X,Y) is:

BRDES(X,Y) = $F\Delta HR^{d}$(X,Y) * MH^{s}(X,Y)

where: MH^{s}(X,Y) is the minimum available margin ratio for steady state DNB at the limiting conditions at assembly location (X,Y).

The reactor core is ″operating as designed″ if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is ″operating as designed″ if the following relationship is satisfied:

$F\Delta HR^{M}$(X,Y) ≤ BHNOM(X,Y)

where: BHNOM(X,Y) is the nominal design radial peaking factor for an assembly at core location (X,Y) increased by an allowance for the expected deviation between the measured and predicted design power distribution.

BASES

APPLICABLE SAFETY ANALYSES

Limits on ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits: 1

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm ~~[~~(Ref. 1~~]~~), and 3 8

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum ~~DNBR to the 95/95 DNB criterion of [1.3] using the [W3] CHF~~ local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB. 1

The allowable ~~$F^N_{\Delta H}$ limit~~ limits, $F_{\Delta H}$ min margin and $f_1(\Delta I)$ min margin, increase~~s~~ with decreasing power level. This functionality in ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ as a function of power level defined by the COLR limit equation. 1

The LOCA safety analysis indirectly models ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F_Q(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature ~~[~~(Ref. 3~~]~~). 3 8

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

(QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ~~$(F^N_{\Delta H})$~~ $F_{\Delta H}(X,Y)$," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F_Q(~~Z~~ X, Y, Z))."

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ and F_Q(~~Z~~ X, Y, Z) are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

LCO

~~$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.~~

INSERT 2

~~The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.~~

~~The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.~~

~~A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of is $F^N_{\Delta H}$ allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.~~

2

1

APPLICABILITY

The ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X,Y)$ in these modes.

1

2 **INSERT 2**

SII

The LCO states that $F_{\Delta H}(X,Y)$ shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured $F_{\Delta H}(X,Y)$ to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing $F\Delta HR^M(X,Y)$ to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the $F_{\Delta H}$ min margin is ≥ 0 and $f_1(\Delta I)$ min margin ≥ 0 the LCO is satisfied.

$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1

B 3.2.2

BASES INSERT 3 2

ACTIONS ~~A.1.1~~

~~With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.~~ 4

Condition A is modified by a Note that requires that Required Actions A.~~2~~ 3 and A.~~3~~ 5 must be completed whenever Condition A is entered. ~~Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.~~ 4

INSERT 4

However, if power is reduced below 50% RTP, Required Action A.~~3~~ 5 requires that another determination of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin must be ~~done~~ verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. ~~In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.~~ 4 1 5

A.1~~.2.1~~ and A.~~1.2.~~2

If the value of ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER ~~to < 50% RTP~~ INSERT 5 in accordance with Required Action A.1~~.2.1~~ and reduce the Power Range Neutron Flux - High ~~to ≤ 55% RTP~~ INSERT 6 in accordance with Required Action A.~~1.2.~~2. Reducing RTP ~~to < 50% RTP~~ INSERT 7 increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of ~~4~~ 2 hours for Required Action A.1~~.2.1 is consistent with those allowed for in Required Action A.1.1 and~~ provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. ~~The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive~~. 1 4 allowable 3 3

2 **INSERT 3**

The % $F_{\Delta H}$ margin is based on the relationship between $F\Delta HR^{M}(X,Y)$ and the limit, BHDES (X,Y), as follows:

$$\%\ F_{\Delta H}\ \text{Margin} = \left(1 - \frac{F\Delta HR^{M}(X,\ Y)}{BHDES(X,\ Y)}\right) x\ 100\%$$

If the reactor core is ″operating as designed″, then $F\Delta HR^{M}(X,Y)$ is less than BHDES (X,Y) and calculation of %$F_{\Delta H}$ margin is not required. If the %$F_{\Delta H}$ margin is less than zero, then $F\Delta HR^{M}(X,Y)$ is greater than BHDES (X, Y) and the $F_{\Delta H}(X,Y)$ limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial conditions DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The $F_{\Delta H}$ min margin is the minimum for all core locations examined.

4 **INSERT 4**

If $F_{\Delta H}$ min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance of Required Actions A.3 and A.5 must be met.

4 **INSERT 5**

from RTP by at least RRH % (where RRH = Thermal power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit) multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 6**

trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 7**

by at least RRH% multiplied times the $F_{\Delta H}$ min margin

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$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1

B 3.2.2

BASES

ACTIONS (continued)

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.~~1.2.~~2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System. 4

A.~~2~~ 3

SII

Once the allowable power level has been reduced ~~to ≤ 50% RTP~~ INSERT 8 per Required Action A.1~~.2.1~~, an incore flux map (SR 3.2.2.1) must be obtained and the ~~measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit~~ $F_{\Delta H}$ min margin is verified ≥ 0 at the lower power level. The unit is provided ~~20~~ 22 additional hours to perform this task over and above the ~~4~~ 2 hours allowed by ~~either Action A.1.1 or~~ Action A.1~~.2.1~~. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin. 4 1 3 1

INSERT 9 4

A.~~3~~ 5 4

Verification that ~~$F^N_{\Delta H}$ is within its specified limits~~ $F_{\Delta H}$ min margin is ~~>~~ ≥ 0 after an out of limit occurrence ensures that the cause that led to the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$ min margin limit is ~~within the LCO limits~~ ~~>~~ ≥ 0 prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. 1

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

INSERT 10 2

~~B~~ C.1 4

When Required Actions A.1~~.1~~ through A.~~3~~ 5, and B.1, cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

---

4 **INSERT 8**

by at least RRH% multiplied times the $F_{\Delta H}$ min margin

4 **INSERT 9**

A.4

If the value of $F\Delta HR^{M}(X,Y)$ is not restored to within its specified limit, Overtemperarture ΔT K1 (OTΔT K1) term is required to be reduced by at least TRH multiplied times the $F_{\Delta H}$ min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

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2 **INSERT 10**

B.1

The %$f_1(\Delta I)$ margin is based on the relationship between $F\Delta HR^M(X,Y)$ and the limit, BRDES (X,Y), as follows:

$$\% \ f_1(\Delta I) Margin = \left(1 - \frac{F\Delta \, HR^M(X,Y)}{BRDES(X,Y)}\right) x \ 100\%$$

If the reactor core is ″operating as designed″, then $F\Delta HR^M(X,Y)$ is less than BRDES (X,Y) and calculation of %$f_1(\Delta I)$ margin is not required. If the %$f_1(\Delta I)$ margin is less than zero, then $F\Delta HR^M(X,Y)$ is greater than BRDES (X, Y) and the OTΔT setpoint limits may not be adequate to prevent exceeding DNB requirements.

BRDES (X,Y) represents the maximum allowable design radial peaking factors which ensure that the steady state DNBR limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties

Required Action B.1 requires the reduction of the OTΔT K1 term by at least TRH multiplied by the $f_1(\Delta I)$ min margin. TRH is the amount of OTΔT K1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR. Completing Required Action B.1 within the allowed Completion Time of 48 hours, restricts $F_{\Delta H}(X,Y)$ such that even if a transient occurred, DNB requirements are met. The $f_1(\Delta I)$ min margin is the minimum % of $f_1(\Delta I)$ margin for all core locations examined.

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$F_{\Delta H}(X,Y)$ ~~$F^N_{\Delta H}$~~ 1
B 3.2.2

BASES

SURVEILLANCE REQUIREMENTS

INSERT 11 4

~~SR 3.2.2.1~~

~~The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.~~

~~After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.~~ 1

~~[The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.~~ 6

~~OR~~

INSERT 12 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~-----]~~

REFERENCES

1. Regulatory Guide 1.77, Rev. ~~[~~0~~]~~, May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.

4. BAW-10163P-A, Revision 0,"Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.

KNH-019

4 **INSERT 11**

SR 3.2.2.1 and SR 3.2.2.2 are modified by a Note. It states that, "Not required to be performed until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained." SR 3.2.2.1 and SR 3.2.2.2 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level. These surveillances could not be satisfactorily performed if the requirement for performance of the Surveillances was included in MODE 2 prior to entering MODE 1.

In a reload core, $F_{\Delta H}^{M}(X,Y)$ could not have previously been measured, therefore, there is a Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_{\Delta H}^{M}(X,Y)$ is made at a lower power level at which adequate margin is available before going to 100% RTP.

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4 **INSERT 12**

SR 3.2.2.1 and SR 3.2.2.2

In addition to ensuring via Surveillance that the nuclear enthalpy rise hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in $F^{M}_{\Delta H}(X,Y)$ for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an $F\Delta HR^{M}(X,Y) > BHNOM(X,Y)$, further consideration is required.

The implications of these extrapolations are considered separately for BHDES(X,Y) and BRDES(X,Y) limits. If the extrapolations of $F^{M}_{\Delta H}(X,Y)$ are unfavorable, additional actions must be taken. These actions are to meet the $F_{\Delta H}(X,Y)$ limit with the last $F^{M}_{\Delta H}(X,Y)$ increased by the appropriate factor specified in the COLR or to evaluate $F^{M}_{\Delta H}(X,Y)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, BASES, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}(X,Y)$)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS 3.2.2 LCO and Action A Bases have been modified to add details associated with the relationship between $F\Delta HR^M(X,Y)$ and BHDES(X,Y) in accordance with NRC Safety Evaluation dated April 27, 1997 (ML013320456).

3. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

4. Changes have been made to be consistent with changes made to the Specification.

5. The ISTS 3.2.2 Bases for A.1.1, 2nd paragraph, contains in part, "Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1." The last paragraph contains a similar statement, " In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours." SQN is deleting the redundant statement in last paragraph.

6. ISTS SR 3.2.2.1 provides two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.2.2.1 under the Surveillance Frequency Control Program.

7. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

8. Editorial change made for clarification.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}(X,Y)$)

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 3

ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

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

A01

ITS ITS 3.2.3

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

A02

in % flux difference units A02

LCO 3.2.3 3.2.1 The ~~indicated AXIAL FLUX DIFFERENCE (~~AFD~~)~~ shall be maintained within the limits specified in the COLR.

≥ M01

Applicability APPLICABILITY: MODE 1 ~~above~~ 50% RATED THERMAL POWER~~*~~

A03

ACTION:

A02

a. With the ~~indicated~~ AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR;

ACTION A

~~1. Either restore the indicated AFD to within the limits within 15 minutes, or~~ L01

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.~~ L02

~~b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.~~ A04

~~*See Special Test Exception 3.10.2~~ A03

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A01

ITS

ITS 3.2.3

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

A02

SR 3.2.3.1

4.2.1.1 The ~~indicated~~ AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

A02

a. Monitoring the ~~indicated~~ AFD for each OPERABLE excore channel ~~at least once per 7 days when the AFD Monitor Alarm is OPERABLE~~. In accordance with the Surveillance Frequency Control Program

LA01

~~b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.~~

L03

A02

LCO 3.2.3 Note

4.2.1.2 The ~~indicated~~ AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits.

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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A01

ITS ITS 3.2.3

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

LCO 3.2.3 3.2.1 The ~~indicated AXIAL FLUX DIFFERENCE (~~AFD~~)~~ in % flux difference units shall be maintained within the limits specified in the COLR. A02

Applicability APPLICABILITY: MODE 1 ~~above~~ ≥ 50% of RATED THERMAL POWER~~*~~. M01 A03

ACTION:

ACTION A

a. With the ~~indicated~~ AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR; A02

~~1. Either restore the indicated AFD to within the limits within 15 minutes, or~~ L01

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.~~ L02

~~b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.~~ A04

~~*See Special Test Exception 3.10.2~~ A03

A01

ITS ITS 3.2.3

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

A02

SR 3.2.3.1

4.2.1.1 The ~~indicated~~ AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

A02

a. Monitoring the ~~indicated~~ AFD for each OPERABLE excore channel ~~at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and~~ In accordance with the Surveillance Frequency Control Program LA01

~~b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.~~

L03

A02

LCO 3.2.3 Note

4.2.1.2 The ~~indicated~~ AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits.

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DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.2.1 states "The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the COLR." CTS 3.2.1 ACTION a provides ACTIONs to take when the indicated AFD is outside the limits. CTS 4.2.1.1 requires a determination that the indicated AFD is within limits. CTS 4.2.1.2 states that the indicated AFD shall be considered outside the limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits. ITS LCO 3.2.3 states "The AFD in % flux difference units shall be maintained within the limits specified in the COLR." ITS LCO 3.2.3 is modified by a Note specifying when AFD is considered to be outside the limits. ITS SR 3.2.3.1 requires verification that AFD is within limits. This changes the CTS by deleting "indicated" and adding "% flux difference units" to the LCO statement.

The purpose of CTS 3.2.1 is to ensure the AFD remains within the limits specified in the COLR. AFD is the difference in normalized flux signals between the top and bottom excore detectors, therefore, this is a presentation change. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.2.1 Applicability contains a footnote (footnote *) which states "See Special Test Exception 3.10.2." ITS 3.2.3 Applicability does not contain this footnote. This changes the CTS by not including Footnote*.

The purpose of Footnote * is to alert the Technical Specification user that a Special Test Exception exists that may modify the Applicability of this Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.2.1 ACTION b states "THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR." ITS 3.2.3 does not contain a similar requirement. This changes the CTS by eliminating a prohibition contained in the CTS.

specify conditions for

This change is acceptable because the requirements have not changed. CTS 3.0.4 and ITS 3.0.4 ~~prohibit~~ entering the MODE of Applicability of a Technical Specification ~~unless the requirements of~~ the LCO ~~are~~ met. CTS 3.2.1 and ITS 3.2.3 are applicable in MODE 1 with THERMAL POWER > 50% RTP (CTS) and ≥ 50 RTP (ITS). Therefore, both the CTS and ITS prohibit exceeding

CSS-006

when

is not

DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

50% RTP without the LCO requirements being met. CTS 3.2.1 ACTION b is duplicative of CTS 3.0.4 and ITS 3.0.4 and its elimination does not make a technical change to the Specification. This change is designated as an administrative change because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.2.1 is applicable in MODE 1 with THERMAL POWER > 50% RTP. ITS 3.2.3 is applicable in MODE 1 with THERMAL POWER ≥ 50% RTP. This changes the CTS by requiring LCO 3.2.3 to be met when THERMAL POWER is equal to 50 % RTP.

The purpose of CTS 3.2.1 is to maintain the AFD within the limits specified in the COLR. When AFD is not within limits, CTS 3.2.1 ACTION a.2, requires reducing THERMAL POWER to less than 50% RTP. This change is acceptable because it aligns the Applicability to the Required Actions. The CTS and ITS Required Action is to reduce THERMAL POWER to less than 50% RTP. When the THERMAL POWER is reduced to this value, it places the core in a condition outside of the Applicability of the LCO. Therefore, changing the Applicability from in MODE 1 with THERMAL POWER > 50% RTP to MODE 1 with THERMAL POWER ≥ 50% RTP has no affect on the LCO. This change is designated as more restrictive because it provides additional requirements to the Applicability.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.2.1.1.a requires monitoring the indicated AFD for each OPERABLE excore channel at least once per 7days. ITS SR 3.2.3.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance

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DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.2.1 ACTION a.1 requires with the AXIAL FLUX DIFFERENCE (AFD) outside of the limits, to restore the indicated AFD to within the limits within 15 minutes. ITS 3.2.3 does not include a Required Action to restore the indicated AFD to within the limits within 15 minutes. This changes the CTS by not including a specific requirement to restore the AFD to within limits.

The purpose of CTS 3.2.1 is to maintain the AFD within the limits specified in the COLR. This change is acceptable because the requirement to restore the AFD to within limits has not changed. ITS 3.2.3 allows a Completion Time of 30 minutes to reduce THERMAL POWER to < 50% RTP. During the time that power is being reduced, AFD can be restored to within limits. Per ITS LCO 3.0.2, if the LCO is met prior to expiration of the Completion Time, completion of the Required Actions is not required. This allowance also is provided in CTS 3.0.2. Therefore, restoration of AFD is always an option and a specific ACTION is not required. This change is designated as less restrictive because additional Completion Time is provided that was not provided in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.2.1 ACTION a.2 states that with the indicated AFD outside of the limits specified in the COLR, reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours. ITS 3.2.3 ACTION A only requires THERMAL POWER to be reduced to less than 50% RTP. This changes the CTS by eliminating the requirement to reduce the Power Range Neutron Flux – High trip setpoints to ≤ 55 % of RTP within the next 4 hours.

The purpose of CTS 3.2.1 ACTION a.2 is to reduce THERMAL POWER to the point at which the LCO is met if AFD is not restored within its limit. With the AFD meeting the Technical Specification requirements, further actions are not required to ensure that the assumptions of the safety analyses are met. Increases in THERMAL POWER are governed by ITS LCO 3.0.4, which requires the LCO to be met prior to entering a MODE or other specified condition in which the LCO applies. Therefore, power increases are prohibited while avoiding the risk of changing Reactor Trip System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.2.1.1.a requires the monitoring of the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE. CTS 4.2.1.1.b

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DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

requires the monitoring and logging of the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging. This changes the CTS by eliminating all AFD Surveillance Frequencies based on the OPERABILITY of the AFD Monitor Alarm.

The purpose of ITS 3.2.3 is to ensure that AFD is within its limit. This change is acceptable because the remaining Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of monitoring AFD when the AFD Monitor Alarm is inoperable is unnecessary as inoperability of the alarm does not increase the probability that AFD is outside of its limit. The AFD Monitor Alarm is for indication only. Its use is not credited in any safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

CTS

AFD ~~(RAOC Methodology)~~
3.2.3~~B~~ 1

3.2 POWER DISTRIBUTION LIMITS

3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~ 1

3.2.1 LCO 3.2.3~~B~~ The AFD in % flux difference units shall be maintained within the limits specified in the COLR. 1

---NOTE---

4.2.1.2 The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

Applicability APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A.2 A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.2.1.1.a SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	~~[7 days~~ ~~OR~~ 3 In accordance with the Surveillance Frequency Control Program ~~]~~ 3

3.2 POWER DISTRIBUTION LIMITS

3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~ 1

3.2.1 LCO 3.2.3~~B~~ The AFD in % flux difference units shall be maintained within the limits specified in the COLR. 1

--NOTE--

4.2.1.2 The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

Applicability APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A.2	A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.2.1.1.a	SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

1. The type of Methodology (Relaxed Axial Offset Control (RAOC)) and the Specification designator "B" are deleted since they are unnecessary (only one AFD Specification is used in the Sequoyah Nuclear (SQN) Plant ITS.) This information is provided in NUREG-1431, Rev. 4.0, to assist in indentifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the Constant Axial Offset Control (CAOC) methodology Specification (ISTS 3.2.3A) is not used and is not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.2.3.1 provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency under the Surveillance Frequency Control Program.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC Methodology)~~ 1

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

~~RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER.~~ The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity. 1

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

~~Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.~~ 1

~~The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.~~ 1

1

BASES

APPLICABLE SAFETY ANALYSES

CSS-007

A Condition 4 event significantly affected by the initial axial power distribution, as indicated by AFD, is the LOCA. A Condition 3 event significantly affected by AFD is the Complete Loss of RCS Flow event. A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power event.

(Ref. 2)

KNH-020

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements. (Ref. 1) KNH-020

~~The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.~~ 1

The limits on the AFD ensure that the Heat Flux Hot Channel Factor (F_Q(X, Y, Z)) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. ~~This ensures that the fuel cladding integrity is maintained for these postulated accidents.~~ ~~The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents.~~ Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints. 2 SII 2 6

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes. KNH-020

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. ~~1 and 2~~ 1 and 3 ~~3~~). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI. 2 6

BASES

LCO (continued)

The AFD limits are provided in the COLR. ~~Figure B 3.2.3B-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.~~ 5 1 INSERT 1

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

~~For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.~~ 1

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. ~~[The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4 3

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1 **INSERT 1**

The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependant envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately ± 5% AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

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1

BASES

REFERENCES

KNH-020

3 ~~1.~~ UFSAR, Section 4.3.2. ~~WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.~~ 2

1 ~~2.~~ BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989. ~~R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.~~ 2

2. ~~3. FSAR, Chapter [15].~~ UFSAR, Chapter 15. 2

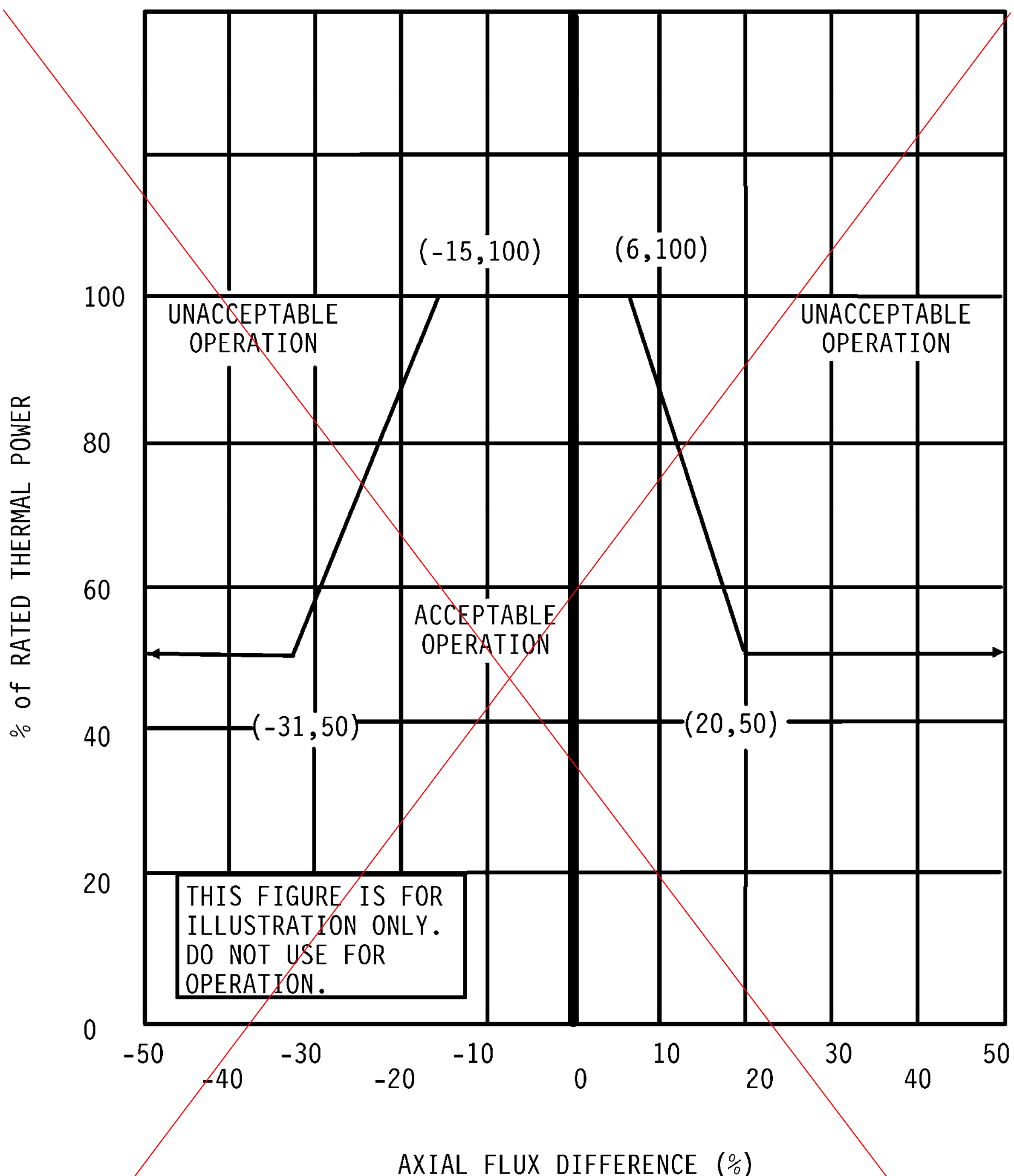

Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

 1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC Methodology)~~ 1

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

~~RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER.~~ The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity. 1

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

~~Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.~~ 1

~~The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.~~ 1

BASES

APPLICABLE SAFETY ANALYSES

CSS-007

A Condition 4 event significantly affected by the initial axial power distribution, as indicated by AFD, is the LOCA. A Condition 3 event significantly affected by AFD is the Complete Loss of RCS Flow event. A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power event.

(Ref. 2)

KNH-020

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements. (Ref. 1) KNH-020

~~The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.~~ 1

The limits on the AFD ensure that the Heat Flux Hot Channel Factor (F_Q(X, Y, Z)) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. ~~This ensures that the fuel cladding integrity is maintained for these postulated accidents.~~ ~~The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents.~~ Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints. 2 SII 2 6

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

KNH-020

1 and 3

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. ~~3~~ ~~1 and 2~~). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI. 2 6

BASES

LCO (continued)

The AFD limits are provided in the COLR. ~~Figure B 3.2.3B-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.~~ INSERT 1

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

~~For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.~~

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. ~~[The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

1 **INSERT 1**

The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependant envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately ± 5% AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

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1

BASES

REFERENCES

3 ~~1~~. ~~WCAP-8403 (nonproprietary),~~ UFSAR, Section 4.3.2. ~~"Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.~~ 2

KNH-020

1 ~~2~~. ~~R. W. Miller et al.,~~ BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989. ~~"Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.~~ 2

2. ~~3. FSAR, Chapter [15].~~ UFSAR, Chapter 15. 2

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1

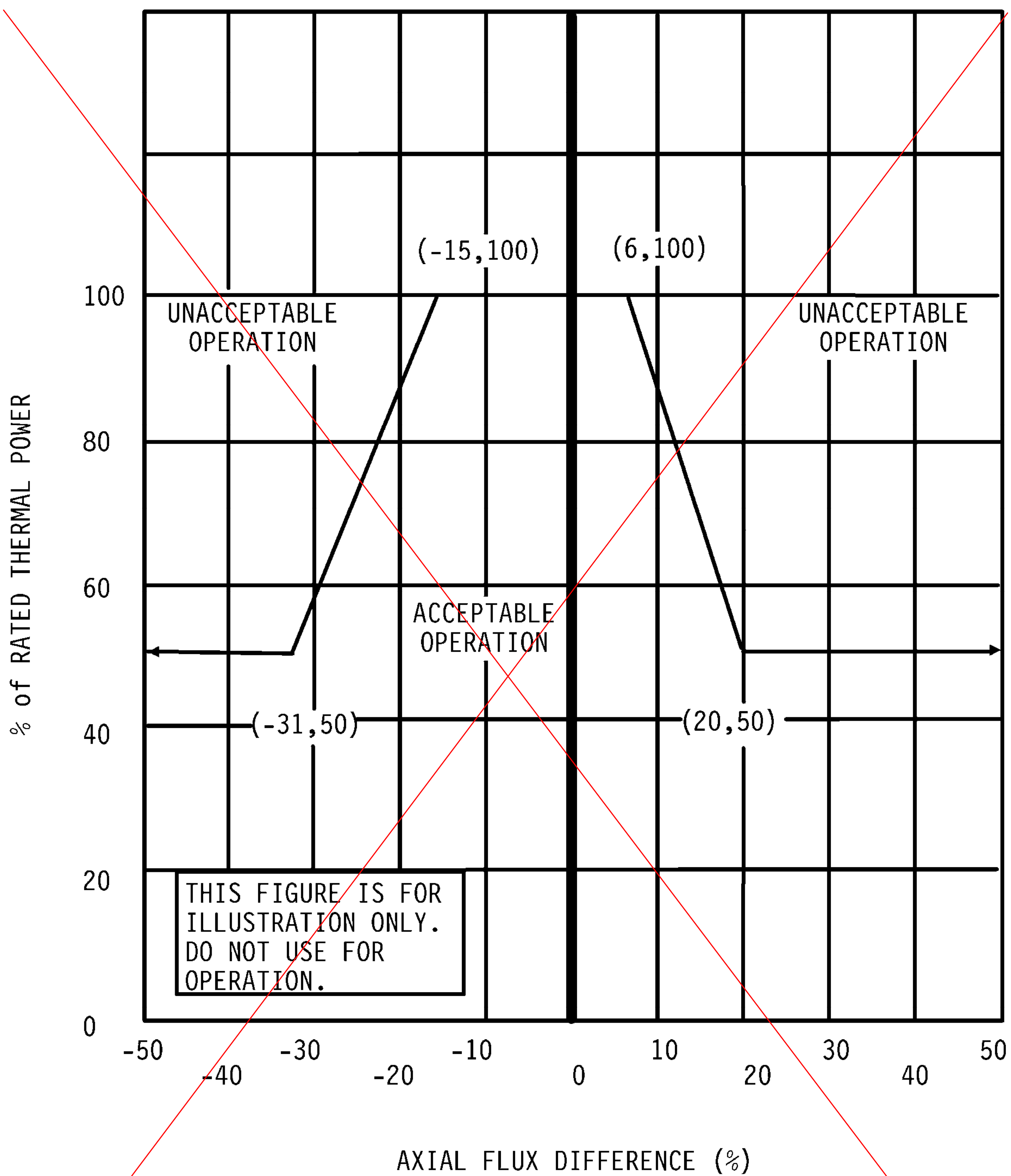

Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

5

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3 BASES, AXIAL FLUX DIFFERENCE (AFD)

1. The type of Methodology (Relaxed Axial Offset Control (RAOC)) and the Specification designator "B" are deleted since they are unnecessary (only one AFD Specification is used in the Sequoyah Nuclear (SQN) Plant ITS.) This information is provided in NUREG-1431, Rev. 4.0, to assist in indentifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the Constant Axial Offset Control (CAOC) methodology Specification (ISTS B 3.2.3A) is not used and is not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.2.3.1 Bases provides two options for controlling the Frequency of Surveillance Requirement. SQN is proposing to control the Surveillance Frequency under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. ISTS 3.2.3 Bases contains Figure B 3.2.3B-1. This Figure is located in the Sequoyah Nuclear Plant (SQN) COLR. Therefore, this figure is not included in the Bases for ITS 3.2.3.

6. Editorial changes made to enhance clarity/consistency.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 4

ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

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

A01

ITS ITS 3.2.4

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

be ≤ A02

LCO 3.2.4 3.2.4 The QUADRANT POWER TILT RATIO shall ~~not exceed~~ 1.02.

A03

Applicability APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER~~*~~

ACTION:

not within limit A04

ACTION A, ACTION B

a. With the QUADRANT POWER TILT RATIO ~~determined to exceed 1.02 but less than or equal to 1.09~~:

12 hours L01

ACTION A

1. Calculate the QUADRANT POWER TILT RATIO at least once per ~~hour~~ until:

A05

~~a)~~ ~~Either the QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or

ACTION B

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

or equal to A06

after each QPTR determination M01

2. Within 2 hours:

a) ~~Either reduce the QUADRANT POWER TILT RATIO to within its limit, or~~ A05

ACTION A

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours~~. L02

~~3.~~ ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours~~.

Add proposed Required Actions A.3, A.4, A.5, and A.6 and proposed ACTION B

~~4.~~ ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

L03

~~*See Special Test Exception 3.10.2.~~

A03

A01

ITS

ITS 3.2.4

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

ACTION A, ACTION B

b. With the QUADRANT POWER TILT RATIO ~~determined to exceed 1.09 due to misalignment of either a shutdown or control rod~~ not within limit: A04

ACTION A

1. Calculate the QUADRANT POWER TILT RATIO at least once per ~~hour~~ 12 hours until: L01

~~a) Either the QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or A05

ACTION B

b) THERMAL POWER is reduced to less than or equal to 50% of RATED THERMAL POWER. A06

ACTION A

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 within ~~30 minutes~~ 2 hours. L04

~~3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours~~.

~~4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~ L03

ACTION A, ACTION B

c. With the QUADRANT POWER TILT RATIO ~~determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod~~ not within limit: A04

ACTION A

1. Calculate the QUADRANT POWER TILT RATIO at least once per ~~hour~~ 12 hours until: L01

~~a) Either the QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or A05

ACTION B

b) THERMAL POWER is reduced to less than or equal to 50% of RATED THERMAL POWER. A06

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A01

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

~~2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for l2 hours or until verified at 95% or greater RATED THERMAL POWER.~~ L03

~~d. With the indicated QUADRANT POWER TILT RATIO not confirmed as required by Surveillance Requirement 4.2.4.2, reduce THERMAL POWER to less than 75 percent RATED THERMAL POWER within 6 hours.~~ L05

~~e. With the QUADRANT POWER TILT RATIO not monitored as required by Surveillance Requirement 4.2.4.1, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 6 hours.~~ L05

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.2.4.1 Notes 1 and 2 L06

SR 3.2.4.1 4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

In accordance with the Surveillance Frequency Control Program LA01

a. Calculating the ratio at ~~least once per 7 days~~ ~~when the alarm is OPERABLE~~.

b. ~~Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable~~. L07

Add proposed SR 3.2.4.2 Note L08

SR 3.2.4.2 4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above
SR 3.2.4.2 Note 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the
SR 3.2.4.2 movable incore detectors ~~to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations or from performance of a full core map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours~~. LA02

In accordance with the Surveillance Frequency Control Program LA01

A01

ITS ITS 3.2.4

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.2.4 3.2.4 The QUADRANT POWER TILT RATIO shall ~~not exceed~~ be ≤ 1.02. A02

Applicability APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER~~*~~ A03

ACTION:

ACTION A, ACTION B a. With the QUADRANT POWER TILT RATIO ~~determined to exceed 1.02 but less than or equal to 1.09~~ not within limit: A04

ACTION A 1. Calculate the QUARANT POWER TILT RATIO at least once per ~~hour~~ 12 hours until either: L01

~~a) The QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or A05

ACTION B b) THERMAL POWER is reduced to less than or equal to 50% of RATED THERMAL POWER. A06

2. Within 2 hours after each QPTR determination ~~either~~: M01 A05

a) ~~Reduce the QUADRANT POWER TILT RATIO to within its limit, or~~

ACTION A b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.~~ L02

~~3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

Add proposed Required Actions A.3, A.4, A.5, and A.6 and proposed ACTION B

~~4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~ L03

~~* See Special Test Exception 3.10.2.~~ A03

A01

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

ACTION A, ACTION B

b. With the QUADRANT POWER TILT RATIO ~~determined to exceed~~ not within limit ~~1.09 due to misalignment of either a shutdown or control rod~~: A04

ACTION A

1. Calculate the QUADRANT POWER TILT RATIO at least once per ~~hour~~ 12 hours until ~~either~~: L01

~~a) The QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or A05

ACTION B

b) THERMAL POWER is reduced to less than or equal to 50% of RATED THERMAL POWER. A06

ACTION A

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 within ~~30 minutes~~ 2 hours. L04

~~3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

L03

ACTION A, ACTION B

c. With the QUADRANT POWER TILT RATIO ~~determined to exceed~~ not within limit ~~1.09 due to causes other than the misalignment of either a shutdown or control rod~~: A04

ACTION A

1. Calculate the QUADRANT POWER TILT RATIO at least once per ~~hour~~ 12 hours until ~~either~~: L01

~~a) The QUADRANT POWER TILT RATIO is reduced to within its limit,~~ or A05

ACTION B

b) THERMAL POWER is reduced to less than or equal to 50% of RATED THERMAL POWER. A06

A01

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

~~2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.~~

L03

~~d. With the indicated QUADRANT POWER TILT RATIO not confirmed as required by Surveillance Requirement 4.2.4.2, reduce THERMAL POWER to less than 75 percent RATED THERMAL POWER within 6 hours.~~

L05

~~e. With the QUADRANT POWER TILT RATIO not monitored as required by Surveillance Requirement 4.2.4.1, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 6 hours.~~

L05

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.2.4.1 Notes 1 and 2 — L06

SR 3.2.4.1

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

In accordance with the Surveillance Frequency Control Program — LA01

a. Calculating the ratio ~~at least once per 7 days~~ when the alarm is OPERABLE.

b. ~~Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.~~

L07

Add proposed SR 3.2.4.2 Note — L08

SR 3.2.4.2

SR 3.2.4.2 Note

SR 3.2.4.2

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors ~~to confirm that the normalized symmetric power distribution, obtained from 4 pairs of symmetric thimble locations or from performance of a full core map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.~~

LA02

In accordance with the Surveillance Frequency Control Program — LA01

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.2.4 states "The QUADRANT POWER TILT RATIO shall not exceed 1.02." ITS LCO 3.2.4 states "The QPTR shall be ≤ 1.02. This changes the CTS by requiring the QPTR to be less than or equal to 1.02.

This change is acceptable because nothing has changed. This is a presentation change for clarity. Stating that the QPTR shall be less than or equal to 1.02 is clearer than stating that it shall not exceed. This change is designated as an administrative change because it does not result in a technical change to the CTS.

A03 CTS 3.2.4 Applicability contains a footnote (footnote *) that states "See Special Test Exceptions 3.10.2." ITS 3.2.4 Applicability does not contain this footnote. This changes the CTS by not including the footnote reference.

The purpose of CTS 3.2.4 footnote * is to alert the user that a Special Test Exception exists which may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as an administrative change since it does not result in a technical change to the CTS.

A04 CTS 3.2.4 ACTION a states "With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09." CTS 3.2.4 ACTION b states "With the QUADRANT POWER TILT RATIO determined to exceed 1.09 resulting from misalignment of either a shutdown or control rod." CTS 3.2.4 ACTION c states "With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod." ITS 3.2.4 ACTION A states "QPTR not within limit." This changes the CTS by specifying that action must be taken when the QPTR is not within limits. (See DOCS L02, L03, and L04 for changes to the compensatory measures.)

The purpose of CTS 3.2.4 is to provide compensatory actions when the QPTR exceeds 1.02. ITS 3.2.4 continues to provide compensatory actions when the QPTR exceeds 1.02. This change is a presentation change. This change is designated as an administrative change since it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

A05 CTS 3.2.4 ACTION a.1.a) states that with QPTR greater than 1.02 and less than or equal to 1.09, calculate the QUADRANT POWER TILT RATIO at least once per hour until either QUADRANT POWER TILT RATIO is reduced to within its limit or THERMAL POWER is reduced to less than 50% of RTP. CTS 3.2.4 ACTION a.2.a) states within 2 hours, either QUADRANT POWER TILT RATIO is reduced to within its limit or reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours. CTS 3.2.4 ACTION b.1.a) states that with QPTR greater than 1.09 due to misalignment of either a shutdown or control rod, calculate the QUADRANT POWER TILT RATIO at least once per hour until either QUADRANT POWER TILT RATIO is reduced to within its limit or THEMAL POWER is reduced to less than 50% of RTP. CTS 3.2.4 ACTION c.1.a) states that with QPTR greater than 1.09 due to causes other than the misalignment of either a shutdown or control rod, calculate the QUADRANT POWER TILT RATIO at least once per hour until either QUADRANT POWER TILT RATIO is reduced to within its limit or THERMAL POWER is reduced to less than 50% of RTP. ITS 3.2.4 does not contain a Required Action stating QPTR must be reduced to within its limit. This changes the CTS by not specifically stating that the restoration of QUADRANT POWER TILT RATIO is required.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action. The convention in the ITS is to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as an administrative change since it does not result in technical changes to the CTS.

A06 CTS 3.2.4 LCO APPLICABLITY is MODE 1 above 50% RTP. CTS 3.2.4 ACTION a.1.b, ACTION b.1.b and ACTION c.1.b state, in part, to calculate the QUADRANT POWER TILT RATIO at least once per hour until either QUADRANT POWER TILT RATIO is reduced to within limit, or THERMAL POWER is reduced to less than 50% of RTP. ITS 3.2.4 LCO APPLICABILITY is MODE 1 with THERMAL POWER >50% RTP. ITS 3.2.4 CONDITION B states that when the Required Action and associated Completion Time are not met to reduce THERMAL POWER to ≤ 50% RTP. This changes the CTS requirement of reducing power and exiting the MODE of APPLICABILITY to a value of < 50% RTP and allow stopping at a value of 50% RTP.

This change is acceptable because the technical requirements have not changed. LCO 3.0.2 states that that when a Required Action to restore variables within limits is not met, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. In this case, both CTS and ITS require a reduction of power to exit the MODE of APPLICABILITY when compliance with the LCO is not met within the prescribed amount of time. Once the MODE of APPLICABILITY for LCO 3.2.4 is exited(>50%), the new power level(50%) is no longer controlled by this specification. This change is designated as an administrative change since it does not result in technical changes to CTS LCO 3.2.4.

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

MORE RESTRICTIVE CHANGES

M01 CTS 3.2.4 ACTION a.2.b states in part, within 2 hours, reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02. ITS 3.2.4 Required Action A.1 has a similar requirement to reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.02. The Completion Time for ITS 3.2.4 Required Action A.1 is 2 hours after each QPTR determination. This changes the CTS by specifically requiring a power reduction, if applicable, after each QPTR determination.

The purpose CTS 3.2.4 ACTION a.2.b is to commence a power level reduction to ensure that core power distributions that violate fuel design criteria are minimized. The maximum allowable power level initially determined by ITS 3.2.4 Required Action A.1 may be affected by subsequent determinations of QPTR. However, any increases in QPTR would require additional power reductions within 2 hours of each QPTR determination, if necessary to comply with the decreased maximum allowable power level. This change is designated as more restrictive because it adds required actions to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.2.4.1 states, in part, the QPTR shall be determined at least once per 7 days by calculating the ratio. CTS 4.2.4.2 states, in part, the QPTR shall be determined, at least once per 12 hours, by using the movable incore detectors. ITS SR 3.2.4.1 and SR 3.2.4.2 require similar Surveillances and specify the periodic Frequencies as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.2.4.2 states, in part, that the QPTR shall be determined to be within the limit by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations or from performance of a full core map, is consistent with the indicated QUADRANT POWER TILT RATIO. ITS SR 3.2.4.2 requires verifying QPTR is within limit using the movable incore detectors. This changes the CTS by moving the procedural details for meeting the Surveillance to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide protection of public health and safety. The ITS still retains the requirement that the QPTR is verified to be within the limits using the movable incore detectors. The details relating to system design do not need to appear in the specification in order for the requirement to apply. Additionally, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.4 ACTIONS a.1, b.1, and c.1 require calculating the QPTR at least once per hour. ITS 3.2.4 ACTION A (Required Action A.2 and associated Completion Time) require, in part, that when the QPTR is not within limit to determine QPTR once per 12 hours. This changes the CTS by requiring the determination of QPTR to be done once per 12 hours instead of once per hour.

The purpose of CTS 3.2.4 ACTIONS a.1, b.1, and c.1 is to verify QPTR until it is brought to within limit or reactor power has been lowered to less than or equal to 50% RTP. This action is taken because with the QPTR not within limit, the core power distribution is not within the analyzed assumptions, and critical parameters such as F_Q (X, Y, Z) and $F_{\Delta H}$ (X,Y) may not be within their limits. In addition to ITS 3.2.4 Required Action A.2 Completion Time the other Required Actions and associated Completion Times of Condition A are consistent with safe operation, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. In addition to reducing reactor power by greater than or equal to 3% for each 1% QPTR exceeds 1.02, ITS 3.2.4 requires a determination of QPTR once per 12 hours. Additionally, ITS 3.2.4 requires

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DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

measurement of F_Q (X, Y, Z) and $F_{\Delta H}$ (X,Y) within 24 hours and every 7 days thereafter to verify that those parameters are within limit. Furthermore, ITS 3.2.4 requires the safety analyses to be reevaluated to ensure that the results remain valid. Assuming that these actions are successful, ITS 3.2.4 allows indefinite operation with QPTR out of its limit and allows the excore nuclear detectors to be normalized to eliminate the indicated QPTR. This ensures the core is operated within the safety analyses. This change is designated as less restrictive because less stringent Completion Times are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 ACTION a.2.b) requires that when QPTR is in excess of 1.02 but less than or equal to 1.09, to reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours. ITS 3.2.4 Required Action A.1 includes the requirement to reduce the THERMAL POWER, but does not include a requirement to reduce the Power Range Neutron Flux-High Trip Setpoints. This changes the CTS by eliminating the requirement to reduce the Power Range Neutron Flux-High Trip Setpoints.

The purpose of CTS 3.2.4 ACTION a.2.b) is to reduce THERMAL POWER to increase the margin to the core power distribution limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while provided time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. With THERMAL POWER reduced by 3% from RTP for each 1% QPTR is greater than 1.02, further actions are not required to ensure that THERMAL POWER is not increased. Power increases are administratively prohibited by the Technical Specification while avoiding the risk of changing Reactor Trip System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 ACTION a.3 states "Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours." CTS 3.2.4 ACTION b.3 and b.4 contain the same compensatory actions as CTS ACTION a.3 but requires the QPTR to be within limits within 2 hours. CTS 3.2.4 ACTIONS a.4, b.4, and c.3 state "Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER."

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

ITS 3.2.4 Required Action A.3 requires performance of SR 3.2.1.1, SR 3.2.1.2, SR 3.2.1.3, SR 3.2.2.1, SR 3.2.2.2 within 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 and once per 7 days thereafter. ITS 3.2.4 Required Action A.4 requires reevaluation of the safety analyses and confirmation that the results remain valid for duration of operation under this condition prior to increasing THERMAL POWER above the limit of Required Action A.1. ITS 3.2.4 Required Action A.5 requires normalization of excore detectors to restore QPTR to within limit prior to increasing THERMAL POWER above the limit of Required Action A.1. ITS 3.2.4 Required Action A.6 requires performance of SR 3.2.1.1, SR 3.2.1.2, SR 3.2.1.3, SR 3.2.2.1, SR 3.2.2.2 within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. Additionally, ITS 3.2.4 Required Action A.5 contains two Notes and ITS 3.2.4 Required Action A.6 contains one Note. ITS 3.2.4 Required Action A.5 Note 1 states "Perform Required Action A.5 only after Required Action A.4 is completed." ITS 3.2.4 Required Action A.5 Note 2 states "Required Action A.6 shall be completed whenever Required Action A.5 is performed." ITS 3.2.4 Required Action A.6 Note states "Perform Required Action A.6 only after Required Action A.5 is completed." Furthermore, ITS 3.2.4 ACTION B states that with a Required Action and associated Completion Time (of Condition A) not met, reduce THERMAL POWER to ≤ 50% RTP within 4 hours. This changes the CTS by eliminating requirements to be ≤ 50% RTP within a specified time of exceeding the LCO and substituting compensatory measures in ITS 3.2.4 ACTION A, which if not met, results in a reduction in power per ITS 3.2.4 ACTION B.

The purpose of the CTS actions is to lower reactor power to less than 50% when QPTR is not within its limit and cannot be restored to within its limit within a reasonable time period. In addition, the Power Range Neutron Flux-High Trip setpoints are reduced to ≤ 55% to ensure that reactor power is not inadvertently increased without QPTR within its limit. This action is taken because with QPTR not within limit, the core power distribution is not within the analyzed assumptions, and critical parameters such as F_Q (X, Y, Z) and $F_{\Delta H}$ (X,Y) may not be within their limits. A QPTR not within limit may not be an unacceptable condition if the critical core parameters such as F_Q (X, Y, Z) and $F_{\Delta H}$ (X,Y) are within their limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while provided time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. ITS 3.2.4 requires measurement of F_Q (X, Y, Z) and $F_{\Delta H}$ (X,Y) within 24 hours and every 7 days thereafter to verify that those parameters are within limit. In addition, ITS 3.2.4 requires the safety analyses to be reevaluated to ensure that the results remain valid. Assuming that these actions are successful, ITS 3.2.4 allows indefinite operation with QPTR out of its limit and allows the excore nuclear detectors to be normalized to eliminate the indicated QPTR. This ensures the core is operated within the safety analyses. This change is designated as less restrictive because less

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DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

stringent Required Actions are being applied in the ITS than were applied in the CTS.

L04 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.4 ACTION b.2, applies when QPTR is greater than 1.09 due to misalignment of either a shutdown or control rod, requires a THERMAL POWER reduction from RATED THERMAL POWER for each 1% of indicated QPTR in excess of 1.02 within 30 minutes. ITS 3.2.4 Required Action A.1 requires a THERMAL POWER reduction of 3% from RTP for each 1% QPTR exceeds 1.02 within 2 hours. This changes the CTS by allowing 2 hours to perform the required power reduction.

The purpose of CTS 3.2.4 is to provide appropriate compensatory actions for QPTR greater than that assumed in the safety analyses. This change is acceptable because the completion Time is consistent with safe operation under the specified Condition, considering other indications available to the operator, a reasonable time for restoring compliance with the LCO, and the low probability of a DBA occurring during the restoration period. Under the ITS, a QPTR of 1.09 would require THERMAL POWER to be reduced to ≤ 79% RTP. This will provide sufficient thermal margin to account for the radial power distribution. In addition, the 2 hour time limit is consistent with the CTS time allowed when QPTR is > 1.02 but ≤ 1.09. This change is designated as less restrictive because additional time is allowed to decrease power than was allowed in the CTS.

L05 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 ACTION d states "With the indicated QUADRANT POWER TILT RATIO not confirmed as required by Surveillance Requirement 4.2.4.2, reduce THERMAL POWER to less than 75 percent RATED THERMAL POWER within 6 hours." CTS 3.2.4 ACTION e states "With the QUADRANT POWER TILT RATIO not monitored as required by Surveillance Requirement 4.2.4.1, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 6 hours." ITS 3.2.4 does not contain these ACTIONS. This changes the CTS by not requiring RTP to be reduced to less than 75 percent, within 6 hours, when the QPTR is not confirmed and not requiring RTP to be reduced to less than 50 percent, within 6 hours, when the QPTR is not monitored.

The purpose of CTS 3.2.4 ACTIONs d and e is to provide compensatory actions to take when Surveillance 4.2.4.1 has not been met or Surveillance 4.2.4.2 have not been performed. ITS 3.2.4 does not contain these ACTIONS since ITS SR 3.0.1 and SR 3.0.3 provide guidance on missed and not performed Surveillances. ITS SR 3.0.1 states, in part, that failure to meet a Surveillance is a failure to meet the LCO. Therefore, the compensatory actions for ITS LCO 3.2.4 would be entered. Additionally, ITS SR 3.0.1 states, in part, that failure to perform a Surveillance shall be failure to meet the LCO, but allows an exception as provided in SR 3.0.3. ITS SR 3.0.3 allows a delayed entry into the LCO to perform the Surveillance. If the Surveillance is not performed in this time period, then the LCO must be declared not met and the compensatory actions for ITS LCO 3.2.4 entered. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

L06 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.2.4.1.a states, in part, that the QPTR shall be determined to be within the limit by calculating the ratio at least once per 7 days. ITS SR 3.2.4.1 requires the same determination, but includes two Notes. ITS SR 3.2.4.1 Note 1 states when the input from one Power Range Neutron Flux channel is inoperable, the remaining three power range channels can be used for calculating QPTR as long as THERMAL POWER is less than or equal to 75% RTP. ITS SR 3.2.4.1 Note 2 states that SR 3.2.4.2 may be performed in lieu of this Surveillance. This changes the CTS by allowing use of three Power Range Neutron Flux channels for calculating the QPTR and by allowing the movable incore detectors to be used to determine QPTR instead of the excore detectors.

The purpose of CTS 4.2.4.1.a is to periodically verify that QPTR is within limit. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are sufficient for verification that the parameters meet the LCO. When one or more Power Range Neutron Flux channels are inoperable, tilt monitoring becomes degraded. With only one Power Range Neutron Flux channel inoperable, QPTR can still be verified by calculation as long as three Power Range Neutron Flux channels are OPERABLE and THERMAL POWER is less than or equal to 75% RTP. The movable incore detector system provides a more accurate indication of QPTR than the excore detectors. In fact, the movable incore detector system is used to calibrate the excore detectors. Therefore, allowing the use of the movable incore detector system or excore detector is appropriate. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L07 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.2.4.1.a states that the QPTR shall be determined to be within the limit by calculating the ratio at least once per 7 days when the alarm is OPERABLE. CTS 4.2.4.1.b states that the QPTR shall be determined to be within the limit by calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable. ITS SR 3.2.4.1 requires verification that the QPTR is within limits every 7 days. This changes the CTS by eliminating the requirement to verify the QPTR more frequently when the QPTR alarm is inoperable.

The purpose of CTS 4.2.4.1.a and 4.2.4.1.b is to periodically verify that the QPTR is within limit. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the frequency of QPTR verification when the QPTR alarm is inoperable is unnecessary as inoperability of the alarm does not increase the probability that the QPTR is outside its limit. The QPTR alarm is for indication only. It use is not credited in any of the safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L08 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.2.4.2 states, in part, that the QPTR shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors. ITS

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

SR 3.2.4.2 requires determination of the QPTR by use of the movable incore detectors. Additionally, ITS SR 3.2.4.2 contains a Note which states "Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP." This changes the CTS by not requiring the Surveillance to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable.

The purpose of CTS 4.2.4.2 is to verify that the QPTR is within limit using the movable incore detectors. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. When one or more Power Range Neutron Flux channels are inoperable, tilt monitoring becomes degraded. Therefore, the movable incore detector system provides a more accurate indication of QPTR than the excore detectors. The ITS SR 3.2.4.2 allowance, for not requiring performance of the Surveillance for 12 hours after input when one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP, is required to allow time for the movable incore detectors to perform the initial measurement of the QPTR before the Surveillance is declared not met. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

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

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

3.2.4 LCO 3.2.4 The QPTR shall be ≤ 1.02.

Applicability APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a, ACTION b, ACTION c DOC M01	A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > ~~1.00~~ 1.02.	2 hours after each QPTR determination	2
		AND		
		A.2 Determine QPTR.	Once per 12 hours	
		AND		
DOC L03		A.3 Perform SR 3.2.1.1, SR 3.2.1.2, ~~and SR 3.2.2.1.~~ SR 3.2.1.3, SR 3.2.2.1 and SR 3.2.2.2.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter	1
		AND		

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3.2.4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L03	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	
DOC L03	A.5 -------------NOTES------------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ------------------------------------- Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	
DOC L03	A.6 ---------------NOTE-------------- Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------------- Perform SR 3.2.1.1, SR 3.2.1.2, ~~and SR 3.2.2.1.~~ SR 3.2.1.3, SR 3.2.2.1 and SR 3.2.2.2.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 1

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3.2.4

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a, ACTION b, ACTION c	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.2.4.1 DOC L06	SR 3.2.4.1	-----------------------------NOTES----------------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

CTS

QPTR
3.2.4

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
4.2.4.2, DOC L08 SR 3.2.4.2	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. --- Verify QPTR is within limit using the movable incore detectors.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

3.2.4

LCO 3.2.4 The QPTR shall be ≤ 1.02.

Applicability

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a, ACTION b, ACTION c DOC M01	A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > ~~1.00~~ 1.02.	2 hours after each QPTR determination	2
		AND		
		A.2 Determine QPTR.	Once per 12 hours	
		AND		
DOC L03		A.3 Perform SR 3.2.1.1, SR 3.2.1.2, ~~and SR 3.2.2.1.~~ SR 3.2.1.3, SR 3.2.2.1 and SR 3.2.2.2.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter	1
		AND		

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QPTR
3.2.4

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
DOC L03	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
DOC L03	A.5	-------------NOTES------------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ------------------------------------- Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
DOC L03	A.6	---------------NOTE-------------- Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------------- Perform SR 3.2.1.1, SR 3.2.1.2, ~~and SR 3.2.2.1.~~ SR 3.2.1.3, SR 3.2.2.1 and SR 3.2.2.2.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 (1)

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3.2.4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a, ACTION b, ACTION c — B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.2.4.1 DOC L06 — SR 3.2.4.1	------------------------------NOTES------------------------------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3

3

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
4.2.4.2, DOC L08 SR 3.2.4.2	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. -- Verify QPTR is within limit using the movable incore detectors.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

1. Changes are made to be consistent with changes made to Specification 3.2.1 and 3.2.2.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.2.4.1 and SR 3.2.4.2 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and 1

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor (F_Q(X, Y, Z)), the Nuclear Enthalpy Rise Hot Channel Factor ~~($F^N_{\Delta H}$)~~ ($F_{\Delta H}$(X, Y)), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits. 1

The QPTR limits ensure that ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X, Y) and F_Q(X, Y, Z) remain below their limiting values by preventing an undetected change in the gross radial power distribution. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

In MODE 1, the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X, Y)$ and F_Q(X, Y, Z) limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. 1

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in F_Q(X, Y, Z) and ~~($F^N_{\Delta H}$)~~ $F_{\Delta H}(X, Y)$ is possibly challenged. 1

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X, Y)$ and F_Q(X, Y, Z) LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower. 1

ACTIONS

<u>A.1</u>

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds ~~1.00~~ 1.02 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. 1

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors F_Q(X, Y, Z)~~, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$,~~ and $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of the applicable LCOs of ~~these Surveillances~~ provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

1 1 CSS-008 STET 3 1

A.4

Although $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod 1

BASES

ACTIONS (continued)

malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR is still exceeding ~~has exceeded~~ the 1.02 limit and a re-evaluation of the safety 6
analysis is completed and shows that safety requirements are met, the
excore detectors shall be ~~are~~ normalized to restore QPTR to within limits prior to 6
increasing THERMAL POWER to above the limit of Required Action A.1.
Normalization is accomplished in such a manner that the indicated QPTR
following normalization is near 1.02 ~~1.00~~. This is done to detect any 1
subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR shall not be ~~is not~~ restored to within limits by excore detector normalization until after the re-evaluation of the 2
safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification

BASES

ACTIONS (continued)

X, Y, $F_{\Delta H}(X, Y)$

that $F_Q(Z)$~~, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$,~~ and ~~$F_{\Delta H}^N$~~ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. 1

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ ~~<~~ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems. 1

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. ~~[The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~---]~~ 4

For those causes of ~~QPT~~ QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt. 6

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. ~~[Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.~~ 4

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~---]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 ~~for three and four loop cores~~. 2

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77, Rev ~~[~~0~~]~~, May 1974. 2
3. 10 CFR 50, Appendix A, GDC 26.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and 1

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor (F_Q(X, Y, Z)), the Nuclear Enthalpy Rise Hot Channel Factor ~~($F^N_{\Delta H}$)~~ ($F_{\Delta H}$(X, Y)), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits. 1

The QPTR limits ensure that ~~$F^N_{\Delta H}$~~ $F_{\Delta H}$(X, Y) and F_Q(X, Y, Z) remain below their limiting values by preventing an undetected change in the gross radial power distribution. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

In MODE 1, the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X, Y)$ and F_Q(X, Y, Z) limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. 1

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in F_Q(X, Y, Z) and ~~($F^N_{\Delta H}$)~~ $F_{\Delta H}(X, Y)$ is possibly challenged. 1

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the ~~$F^N_{\Delta H}$~~ $F_{\Delta H}(X, Y)$ and F_Q(X, Y, Z) LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower. 1

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds ~~1.00~~ 1.02 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. 1

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors F_Q(X, Y, Z)~~, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$,~~ and $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of the applicable LCOs of ~~these Surveillances~~ provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

CSS-008

STET

1 1 3 1

A.4

Although $F_{\Delta H}$(X, Y) ~~$F^N_{\Delta H}$~~ and F_Q(X, Y, Z) are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod 1

BASES

ACTIONS (continued)

malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

is still exceeding

If the QPTR ~~has exceeded~~ the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors shall be ~~are~~ normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.02 ~~1.00~~. This is done to detect any subsequent significant changes in QPTR. 6 6 1

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR shall not be ~~is not~~ restored to within limits by excore detector normalization until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions. 2

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification

BASES

ACTIONS (continued)

X, Y, $F_{\Delta H}(X, Y)$

that F_Q(Z)~~, as approximated by F^C_Q(Z) and F^W_Q(Z),~~ and ~~$F^N_{\Delta H}$~~ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. 1

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ~~<~~ ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems. 1

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. ~~[The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~---]~~ 4

For those causes of ~~QPT~~ QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt. 6

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. ~~[Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.~~ 4

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~---]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 ~~for three and four loop cores~~. 2

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77, Rev ~~[~~0~~]~~, May 1974. 2
3. 10 CFR 50, Appendix A, GDC 26.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4 BASES, QUADRANT POWER TILT RATIO (QPTR)

1. Changes are made to be consistent with changes made to the Specification.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS 3.2.4 Bases Required Action A.3 refers to the Required Actions of the referenced Surveillances. There are no Required Actions in the ITS 3.2.1 or ITS 3.2.2 Surveillances. This reference has been corrected to refer to the Required Actions of the applicable LCOs.

4. ISTS SR 3.2.4.1 and SR 3.2.4.2 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Additionally, the Frequency description which is being removed will be included in the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. Typographical/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.4, QUADRANT POWER TILT RATIO

There are no specific No Significant Hazards Considerations for this Specification.

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ENCLOSURE 2

VOLUME 9

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.4
REACTOR COOLANT SYSTEM (RCS)

Revision 0

LIST OF ATTACHMENTS

1. ITS 3.4.1 – RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits
2. ITS 3.4.2 – RCS Minimum Temperature for Criticality
3. ITS 3.4.3 – RCS Pressure and Temperature Limits
4. ITS 3.4.4 – RCS Loops – MODES 1 and 2
5. ITS 3.4.5 – RCS Loops – MODE 3
6. ITS 3.4.6 – RCS Loops – MODE 4
7. ITS 3.4.7 – RCS Loops – MODE 5, Loops Filled
8. ITS 3.4.8 – RCS Loops – MODE 5, Loops Not Filled
9. ITS 3.4.9 – Pressurizer
10. ITS 3.4.10 – Pressurizer Safety Valves
11. ITS 3.4.11 – Pressurizer Power Operated Relief Valves
12. ITS 3.4.12 – Low Temperature Overpressure Protection System
13. ITS 3.4.13 – RCS Operational LEAKAGE
14. ITS 3.4.14 – RCS Pressure Isolation Valve Leakage
15. ITS 3.4.15 – RCS Leakage Detection Instrumentation
16. ITS 3.4.16 – RCS Specific Activity
17. ITS 3.4.17 – Steam Generator (SG) Tube Integrity
18. ISTS Not Adopted

ATTACHMENT 1

ITS 3.4.1, RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits

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

ITS A01 ITS 3.4.1

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

LCO 3.4.1.b a. Reactor Coolant System (RCS) T_{avg}

LCO 3.4.1.a b. Pressurizer Pressure

LCO 3.4.1.c c. RCS Total Flow Rate

Applicability APPLICABILITY: MODE 1

ACTION:

ACTION A With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than or equal to 5% of RATED THERMAL POWER within the next

ACTION B 4 6 hours. A02 L01

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1, SR 3.4.1.2, SR 3.4.1.3 4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits ~~at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program. LA01

SR 3.4.1.4 4.2.5.2 The RCS total flow rate shall be determined by measurement In accordance with the Surveillance Frequency Control Program ~~at least once per 18 months~~.

~~4.2.5.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.~~ LA02

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ITS (A01) ITS 3.4.1

TABLE 3.2-1

DNB PARAMETERS

SII

	PARAMETER	LIMITS ~~4 Loops In Operation~~ (A03)
LCO 3.4.1	Reactor Coolant System T_{avg}	STET ~~≤ 583°F~~
	Pressurizer Pressure	STET ~~≥ 2220 psia*~~
	Reactor Coolant System Total Flow	~~Figure 3.2-1~~

~~as specified in the COLR~~ ~~LA03~~

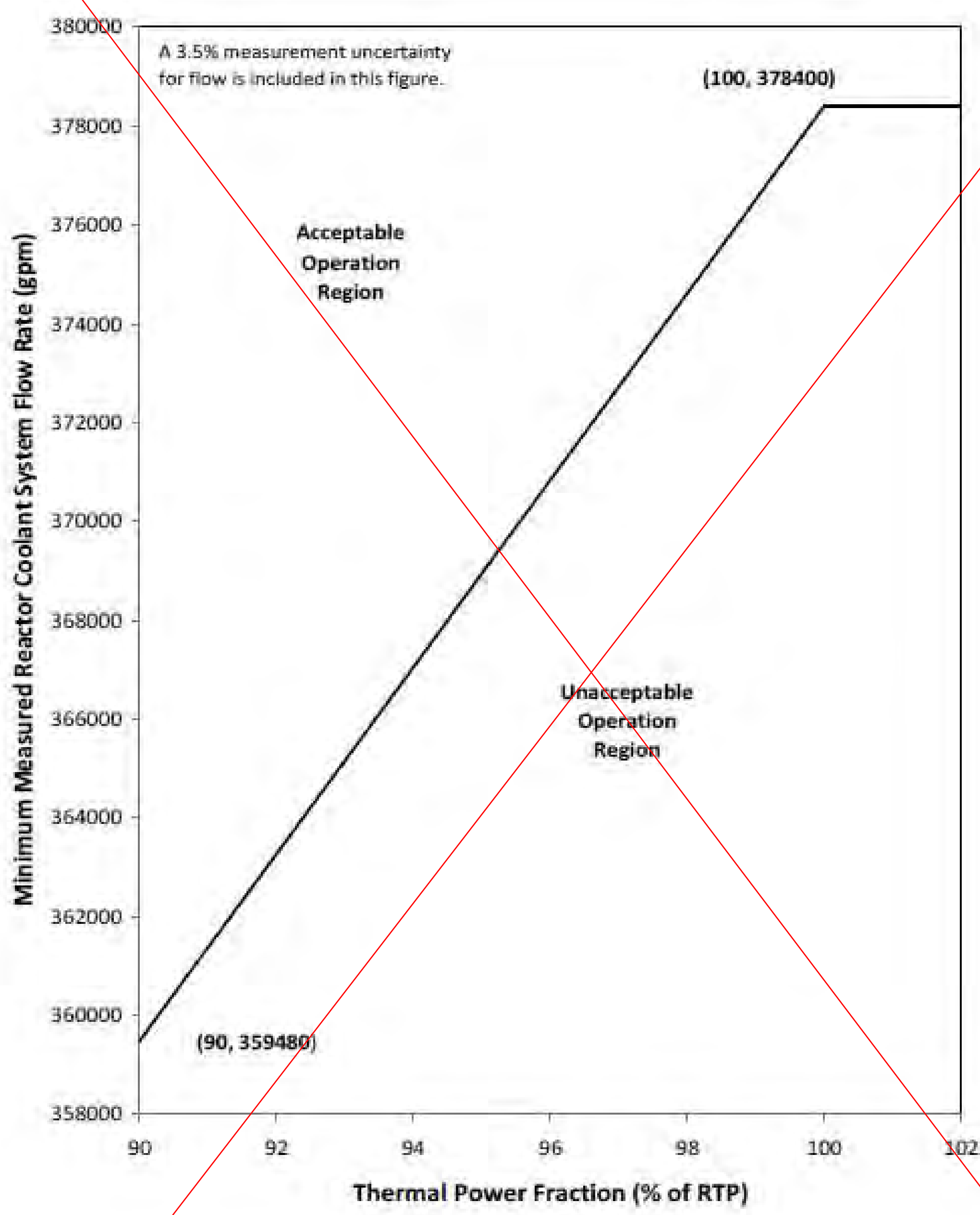
378,400 gpm M01 ~~LA03~~

Applicability Note *Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

ITS A01 ITS 3.4.1

~~Figure 3.2-1 Flow vs. Power for 4 Loops in Operation~~

SII



M01

~~LA03~~

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ITS A01 ITS 3.4.1

POWER DISTRIBUTION LIMITS

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

LCO 3.4.1.b a. Reactor Coolant System T_{avg}.

LCO 3.4.1.a b. Pressurizer Pressure.

LCO 3.4.1.c c. Reactor Coolant System (RCS) Total Flow Rate.

Applicability APPLICABILITY: MODE 1.

ACTION:

ACTION A With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than or equal to 5% of RATED THERMAL POWER within the next

ACTION B ~~4~~ 6 hours.

A02 L01

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1, SR 3.4.1.2, SR 3.4.1.3 4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits ~~at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program.

LA01

SR 3.4.1.4 4.2.5.2 The RCS flow rate shall be determined by measurement ~~at least once per 18 months~~ In accordance with the Surveillance Frequency Control Program.

~~4.2.5.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.~~ LA02

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ITS

A01

ITS 3.4.1

TABLE 3.2-1

DNB PARAMETERS

SII

	PARAMETER	LIMITS ~~4 Loops In Operation~~	
LCO 3.4.1	Reactor Coolant System T_{avg}	STET ~~≤ 583°F~~	A03
	Pressurizer Pressure	STET ~~≥ 2220 psia*~~	~~as specified in the COLR~~ ~~LA03~~
	Reactor Coolant System Flow Rate	~~Figure 3.2-1~~ 378,400 gpm	M01 ~~LA03~~

Applicability Note

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

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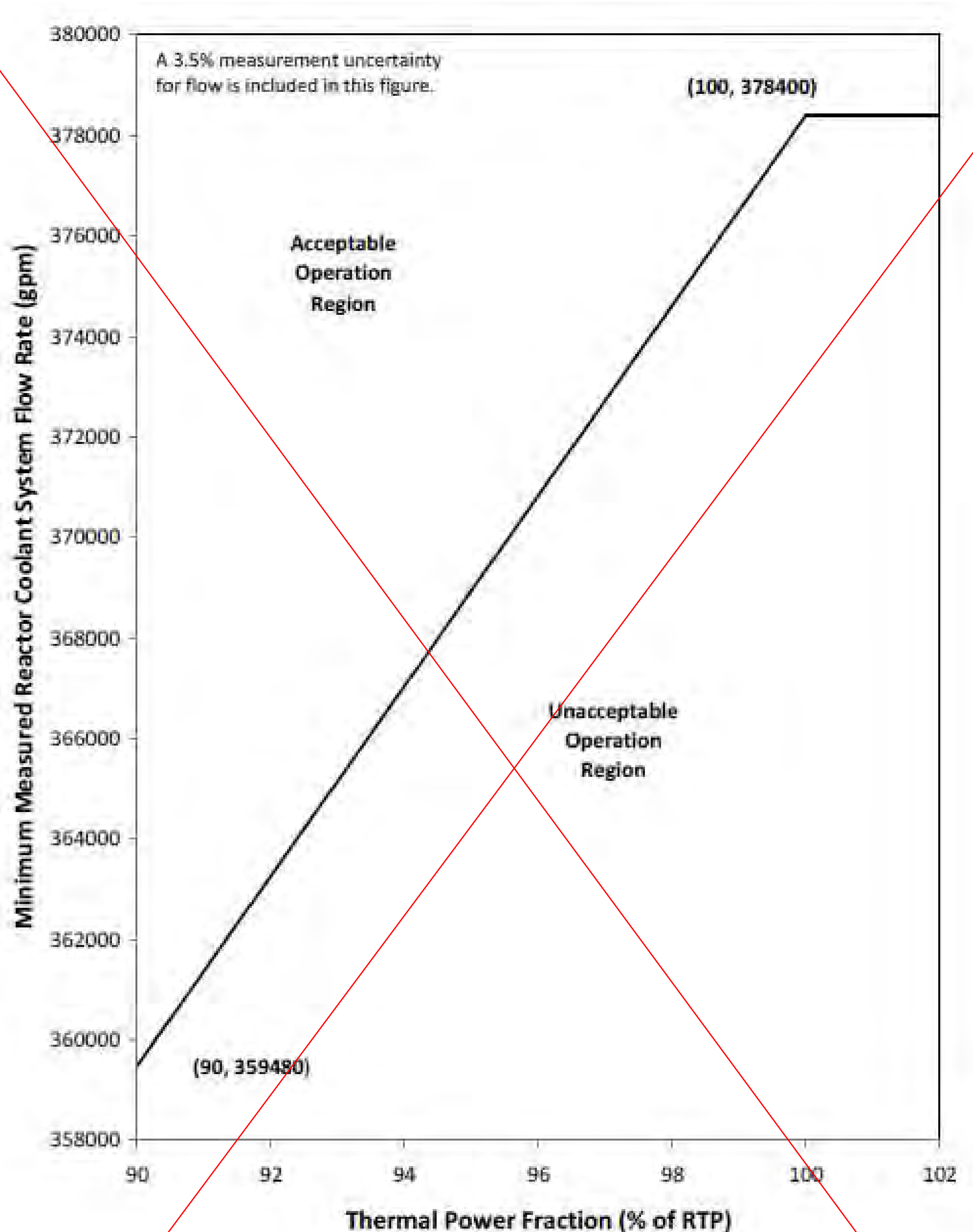
ITS

A01

ITS 3.4.1

~~Figure 3.2-1 Flow vs. Power for 4 Loops in Operation~~

SII



M01

~~LA03~~

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DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.2.5 ACTION requires the unit to reduce THERMAL POWER to "less than" 5% of RATED THERMAL POWER within the next 4 hours if the DNB parameters are not restored to within limit in 2 hours. ITS 3.4.1 ACTION B requires the power reduction to "less than or equal to" 5% RTP (MODE 2) within the next 6 hours if the DNB parameters are not restored to within limit in 2 hours. This changes the CTS by allowing the unit to be at 5% RTP instead of < 5% RTP. The change in the time period to reach 5% RTP is discussed in DOC L01.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.2.5 is applicable in MODE 1, which is greater than 5% RTP. CTS 3.0.1 states that compliance with the LCO is required during the MODES or other conditions specified within the Specification, except that upon failure to meet the LCO, the associated ACTION requirements shall be met. Therefore, the CTS 3.2.5 ACTION to be less than 5% RTP is no longer applicable once the unit enters MODE 2, i.e., at 5% RTP, and the ACTION is exited. As a result, changing the ACTION to "be in MODE 2" results in no operational difference from the CTS Action. This change is designated as administrative as it results in no technical change to the CTS.

A03 CTS Table 3.2-1 contains a column for DNB limits during four loop operation. The ITS does not contain this detail. This changes the CTS by eliminating the detail that the DNB limits apply to four loop operation.

This change is acceptable because the requirements have not changed. Both the ITS and the CTS require all four loops in operation. This change is designated as administrative because it eliminates detail in the CTS that is not necessary.

MORE RESTRICTIVE CHANGES

~~None~~ ← $\boxed{\text{Insert 1}}$ $\boxed{\text{SII}}$

RELOCATED SPECIFICATIONS

None

SII

INSERT 1

M01

CTS Figure 3.2-1 Flow vs. Power for 4 Loops in Operation provides the acceptable Minimum Measured Reactor Coolant System Flow Rate (measured in gallons per minute (gpm)) based on the Thermal Power Fraction (% of Rated Thermal Power (RTP)). The Acceptable Operation Region covers flows from 100% RTP (minimum RCS flow rate 378,400 gpm) to 90% RTP (minimum RCS flow rate 359,480 gpm). ITS LCO 3.4.1 does not include a figure for a minimum acceptable RCS flow rate. ITS LCO 3.4.1 requires RCS total flow rate to be greater than or equal to the required flow for 100% RTP. This changes the CTS by not allowing the RCS total flow rate to be < 378,400 in MODE 1.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for Departure from Nucleate Boiling analyses. Therefore, this change is acceptable because it ensures the RCS flow rate assumed in the minimum departure from nucleate boiling ratio will be met for each transient analyzed. This change is designated as more restrictive because plant operations are more limited by the ITS requirements than the CTS requirements.

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.2.5.1 requires verification that the DNB parameters are within their limits at least once per 12 hours. ITS SR 3.4.1.1, SR 3.4.1.2, and SR 3.4.1.3 require similar Surveillances and specify a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." CTS 4.2.5.2 requires determination of RCS total flow by measurement at least once per 18 months. ITS SR 3.4.1.4 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for the SRs and the Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 4.2.5.3 requires that the indicators which are used to determine RCS flow rate be subjected to a CHANNEL CALIBRATION at least once per 18 months. ITS 3.4.1 does not include this requirement. This changes the CTS by relocating the Surveillance Requirement to the Technical Requirements Manual (TRM).

The removal of requirements for indication-only instrumentation and alarms from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The RCS flow rate indicators are not required to be OPERABLE to determine whether the RCS total flow rate is within limit. The requirement to determine RCS total flow rate remains in the ITS. In addition, the majority of the instrumentation (e.g., sensor) remains in the ITS as part of ITS 3.3.1 (Table 3.3.1-1 Function 10). Also, this change is acceptable because the removed information will be adequately controlled in the TRM. The TRM is incorporated by reference into the UFSAR and any changes to the TRM are made under 10 CFR 50.59, thereby ensuring changes are properly evaluated. This change is designated as a less restrictive removal of detail change because performance requirements for indication-only instrumentation is being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

SII

Not Used

LA03 ~~*(Type 6 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS Table 3.2-1 places limits on RCS average temperature, pressurizer pressure, and RCS total flow rate. ITS LCO 3.4.1 states that the limits on RCS average temperature, pressurizer pressure, and RCS total flow rate shall not exceed the limits specified in the COLR. This changes the CTS by relocating the specific values of RCS average temperature, pressurizer pressure, and RCS total flow rate, that must be confirmed on a cycle-specific basis, to the COLR.~~

~~The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. NRC-approved Topical Report WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for the DNB parameters may be relocated to the COLR as long as the limiting RCS total flow limit is retained in the LCO. The LCO continues to require that the core be operated within the DNB limits. The methodologies used to develop the DNB parameters in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "CORE OPERATING LIMITS REPORT." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.~~

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.5 ACTION requires THERMAL POWER to be reduced to less than 5% of RTP within the next 4 hours if the DNB parameters are not restored to within limit in 2 hours. ITS 3.4.1 ACTION B requires the power reduction to less than or equal to 5% RTP (MODE 2) within the next 6 hours if the DNB parameters are not restored to within limit in 2 hours. This changes the CTS by extending the time for the unit to be placed outside the MODE of Applicability. The change that allows the THERMAL POWER reduction to be only to 5% RTP is discussed in DOC A02.

The purpose of the CTS 3.2.5 ACTION is to limit the time the unit can be outside the DNB parameter limits and remain within the Applicability of the Specification. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The change extends the time the unit is allowed to be outside the DNB parameter limits and be in the Applicability of the Specification. The time extension is from 4 hours to 6 hours. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

SII

LCO 3.2.5, Table 3.2-1, Figure 3.2-1

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

LCO 3.2.5b

a. Pressurizer pressure is ≥ 2220 psia ~~greater than or equal to the limit specified in the COLR,~~ ; 7 1

LCO 3.2.5a

b. RCS average temperature is ≤ 583°F ~~less than or equal to the limit specified in the COLR,~~ ; and 7 1

LCO 3.2.5c

M01

c. RCS total flow rate ≥ ~~[284,000]~~ 378,400 gpm ~~and greater than or equal to the limit specified in the COLR~~. 2 7

Applicability

APPLICABILITY: MODE 1.

Table 3.2.1 Footnote*

--NOTE--
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute ~~or~~ ;

b. THERMAL POWER step > 10% RTP ~~.~~ ;

INSERT 1
-- 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

ACTION (A)

ACTION (B)

CTS 3.4.1

3

INSERT 1

Table 3.2-1
Note *

c. PHYSICS TESTS; or

d. Performance of SR 3.1.3.2.

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SURVEILLANCE REQUIREMENTS

SII

	SURVEILLANCE	FREQUENCY	
4.2.5.1	SR 3.4.1.1 Verify pressurizer pressure is ~~greater than or equal to the limit specified in the COLR~~ ≥ 2220 psia.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 7 5
4.2.5.1	SR 3.4.1.2 Verify RCS average temperature is ~~less than or equal to the limit specified in the COLR~~ ≤ 583°F.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 7 5
4.2.5.1 M01	SR 3.4.1.3 Verify RCS total flow rate is ≥ ~~[284,000]~~ 378,400 gpm ~~and greater than or equal to the limit specified in the COLR~~.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 7 5

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.2.5.2	SR 3.4.1.4	~~-----NOTE-----~~ ~~Not required to be performed until 24 hours after ≥ [90]% RTP.~~ ~~----------~~		4
M01		Verify by measurement ~~precision heat balance~~ that RCS total flow rate is ≥ 378,400 ~~[284,000]~~ gpm ~~and greater than or equal to the limit specified in the COLR~~.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	6 5 2 7 SII 5

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

SII

LCO 3.2.5, Table 3.2-1, Figure 3.2-1

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

LCO 3.2.5b

a. Pressurizer pressure is ~~greater than or equal to the limit specified in the COLR,~~ ≥ 2220 psia ; 7 1

LCO 3.2.5a

b. RCS average temperature is ~~less than or equal to the limit specified in the COLR,~~ ≤ 583°F ; and 7 1

LCO 3.2.5c

M01

c. RCS total flow rate ≥ ~~[284,000]~~ 378,400 gpm ~~and greater than or equal to the limit specified in the COLR~~. 2 7

Applicability

APPLICABILITY: MODE 1.

Table 3.2.1 Footnote*

---NOTE---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute ~~or~~ ;

b. THERMAL POWER step > 10% RTP~~.~~ ;

INSERT 1

--- 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

3

INSERT 1

Table 3.2-1 Note *

c. PHYSICS TESTS; or

d. Performance of SR 3.1.3.2.

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	SII
4.2.5.1	SR 3.4.1.1 Verify pressurizer pressure is ~~greater than or equal to the limit specified in the COLR~~. ≥ 2220 psia	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 7 5
4.2.5.1	SR 3.4.1.2 Verify RCS average temperature is ~~less than or equal to the limit specified in the COLR~~. ≤ 583°F	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 7 5
4.2.5.1 M01	SR 3.4.1.3 Verify RCS total flow rate is ≥ ~~[284,000]~~ 378,400 gpm ~~and greater than or equal to the limit specified in the COLR~~.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 7 5

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.2.5.2	SR 3.4.1.4	~~NOTE~~ ~~Not required to be performed until 24 hours after ≥ [90]% RTP.~~		4
M01		Verify by measurement ~~precision heat balance~~ that RCS total flow rate is ≥ 378,400 ~~[284,000]~~ gpm ~~and greater than or equal to the limit specified in the COLR~~.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	6 5 2 7 SII 5

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

1. The punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made to ISTS 3.4.1 Applicability Note to provide allowances for the pressurizer pressure DNB limit to not apply during PHYSICS TESTS and performance of the moderator temperature coefficient determination (SR 3.1.3.2), as established in NRC Safety Evaluation for Amendments 33/25, Units 1/2, respectively (ML013250071) dated March 29, 1984.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. ISTS SR 3.4.1.1 (ITS SR 3.4.1.1), ISTS SR 3.4.1.2 (ITS SR 3.4.1.2), ISTS SR 3.4.1.3 (ITS SR 3.4.1.3), and ISTS SR 3.4.1.4 (ITS SR 3.4.1.4) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

6. ISTS SR 3.4.1.4 has been changed to reflect current licensing basis. SQN used heat balance RCS flow measurements until the RTD Bypass manifolds were removed, after which problems were experienced due to hot leg streaming. SQN changed licensing basis to allow the measurement of RCS total flow using elbow tap flow differential pressure method.

7. ISTS LCO 3.4.1.a, 3.4.1.b, and 3.4.1.c allow the values for pressurizer pressure, RCS average temperature, and RCS total flow rate to be specified in the COLR. ISTS SR 3.4.1.1, ISTS SR 3.4.1.2, and ISTS SR 3.4.1.3 require verification that the pressurizer pressure, RCS average temperature, and RCS total flow rate meet values specified in the COLR. For ITS 3.4.1 (ISTS 3.4.1), the numerical values for pressurizer pressure, RCS average temperature, and RCS total flow rate are not relocated to the COLR. They are retained in the Specification and reflect the current licensing basis.. SII

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

INSERT 1

The RCS pressure limit is consistent with operation within the nominal operational envelope. ~~Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.~~ A lower pressure will cause the reactor core to approach DNB limits. 1

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

INSERT 2

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. ~~Flow~~ rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits. 1

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or ~~stuck~~ misaligned rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." 1

1 **INSERT 1**

Each pressurizer pressure indication is compared to the limit.

1 **INSERT 2**

Each OPERABLE flow rate indication is compared to the limit. If one or more flow rate indications are unavailable, the remaining flow

BASES

APPLICABLE SAFETY ANALYSES (continued)

SII

The pressurizer pressure limit and RCS average temperature limit ~~specified in the COLR~~ correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty. 3

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. ~~These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However,~~ T the minimum RCS flow~~, usually~~ is based on ~~[~~ the maximum analyzed steam generator tube plugging~~], is retained in the TS LCO~~. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient. 3 1 2

INSERT 3 ~~RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance for no fouling.~~ 1

~~Any fouling that might bias the flow rate measurement greater than the penalty for undetected fouling of the feedwater venturi can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.~~

The numerical values for pressure, temperature, and flow rate specified in the ~~COLR~~ LCO are given for the measurement location and have been adjusted for instrument error. 3

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

1 **INSERT 3**

RCS flow indication calibration must include appropriate considerations for the accuracy of feedwater flow measurement. Sequoyah Nuclear Plant (SQN) can employ either of two methods to measure feedwater flow; an installed Leading Edge Flow Meter (LEFM), or in-line feedwater flow venturis. Unlike the feedwater venturis, the LEFM is not susceptible to fouling during use and possesses a higher accuracy. These attributes make the LEFM the preferred method of measuring feedwater flow as an input to the determination of RCS flow.

In the event the LEFM is unavailable, the feedwater venturis are used to calibrate the RCS flow indicators. However, the calibration assumptions for flow measurement uncertainties is not applicable to the case where the power calorimetric is based on the venturi feedwater flow indication, even if the LEFM is used to correct the venturi feedwater flow indications for the effects of fouling. For those instances where the LEFM is unavailable, SQN Technical Requirements Manual (TRM) specifies the appropriate actions to be taken.

BASES

APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. INSERT 4 These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations. 3

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, ~~as required by Required Action B.1,~~ to restore DNB margin and eliminate the potential for violation of the accident analysis bounds. 4

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

3 **INSERT 4**

Additionally, the limit on pressurizer pressure is not applicable during PHYSICS TESTS and during the performance of SR 3.1.3.2, moderator temperature coefficient (MTC) determination. Measurement of MTC has a high probability of causing a drop in pressure below the specified value, because the reactor coolant system temperature must be dropped several degrees below T_{avg} for an accurate MTC measurement. This results in an associated drop in pressurizer level and in a downswing of pressurizer pressure, making it difficult to maintain pressurizer pressure above the limit.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

[~~Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~] 6

SR 3.4.1.2

[~~Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~] 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

[~~The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~]

5

6

SR 3.4.1.4

an elbow tap differential flow method (Ref. 2)

Measurement of RCS total flow rate by performance of ~~a precision calorimetric heat balance~~ allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. 1

[~~The Frequency of [18] months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~]

5

6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ [90%] RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.~~ 3

REFERENCES 1. U FSAR, ~~Section~~ Chapter ~~[~~15~~]~~. 2 1

2. UFSAR, Section 7.2.2.2.2. 1

SII

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

INSERT 1

The RCS pressure limit is consistent with operation within the nominal operational envelope. ~~Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.~~ A lower pressure will cause the reactor core to approach DNB limits. 1

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

INSERT 2

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. ~~Flow~~ rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits. 1

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

misaligned

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or ~~stuck~~ rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." 1

1 **INSERT 1**

Each pressurizer pressure indication is compared to the limit.

1 **INSERT 2**

Each OPERABLE flow rate indication is compared to the limit. If one or more flow rate indications are unavailable, the remaining flow

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BASES

APPLICABLE SAFETY ANALYSES (continued) SII

The pressurizer pressure limit and RCS average temperature limit ~~specified in the COLR~~ correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty. 3

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. ~~These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However,~~ T the minimum RCS flow~~, usually~~ is based on the [maximum analyzed steam generator tube plugging]~~, is retained in the TS LCO~~. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient. 3 1 2

INSERT 3 ~~RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance for no fouling.~~ 1

~~Any fouling that might bias the flow rate measurement greater than the penalty for undetected fouling of the feedwater venturi can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.~~

The numerical values for pressure, temperature, and flow rate specified in the ~~COLR~~ LCO are given for the measurement location and have been adjusted for instrument error. 3

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

1 **INSERT 3**

RCS flow indication calibration must include appropriate considerations for the accuracy of feedwater flow measurement. Sequoyah Nuclear Plant (SQN) can employ either of two methods to measure feedwater flow; an installed Leading Edge Flow Meter (LEFM), or in-line feedwater flow venturis. Unlike the feedwater venturis, the LEFM is not susceptible to fouling during use and possesses a higher accuracy. These attributes make the LEFM the preferred method of measuring feedwater flow as an input to the determination of RCS flow.

In the event the LEFM is unavailable, the feedwater venturis are used to calibrate the RCS flow indicators. However, the calibration assumptions for flow measurement uncertainties is not applicable to the case where the power calorimetric is based on the venturi feedwater flow indication, even if the LEFM is used to correct the venturi feedwater flow indications for the effects of fouling. For those instances where the LEFM is unavailable, SQN Technical Requirements Manual (TRM) specifies the appropriate actions to be taken.

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BASES

APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. INSERT 4 These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations. 3

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, ~~as required by Required Action B.1,~~ to restore DNB margin and eliminate the potential for violation of the accident analysis bounds. 4

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

3 **INSERT 4**

Additionally, the limit on pressurizer pressure is not applicable during PHYSICS TESTS and during the performance of SR 3.1.3.2, moderator temperature coefficient (MTC) determination. Measurement of MTC has a high probability of causing a drop in pressure below the specified value, because the reactor coolant system temperature must be dropped several degrees below T_{avg} for an accurate MTC measurement. This results in an associated drop in pressurizer level and in a downswing of pressurizer pressure, making it difficult to maintain pressurizer pressure above the limit.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

~~[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

5

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

6

SR 3.4.1.2

~~[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

5

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

[~~The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
]

5

6

SR 3.4.1.4

Measurement of RCS total flow rate by performance of ~~a precision calorimetric heat balance~~ an elbow tap differential flow method (Ref. 2) allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. 1

[~~The Frequency of [18] months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
]

5

6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ [90%] RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.~~ 3

REFERENCES 1. U FSAR, ~~Section~~ Chapter ~~[~~15~~]~~. 2 1

2. UFSAR, Section 7.2.2.2.2. 1

SII

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1 Bases, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made to be consistent with changes made to the Specifications

4. Editorial/grammatical error corrected.

5. ISTS SR 3.4.1.1 through SR 3.4.1.4 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.1.1 through SR 3.4.1.4 under the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 2

ITS 3.4.2, RCS Minimum Temperature for Criticality

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

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.2 3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

Applicability APPLICABILITY: MODES 1 and 2# ~~*~~. A02

ACTION:

ACTION A With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, ~~restore (T_{avg}) to within its limit within I5 minutes or~~ be in ~~HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 within the next ~~15~~ 30 minutes. A03 A04

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F:

~~a. Within 15 minutes prior to achieving reactor criticality, and~~

~~b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System (T_{avg}) is less than 551°F, with the T_{avg} - T_{ref} Deviation Alarm not reset.~~ ~~every 12 hours~~ In accordance with the Surveillance Frequency Control Program L01 LA01

Applicability #With K_{eff} greater than or equal to 1.0.
~~*See Special Test Exception 3.10.3.~~ A02

ITS A01 ITS 3.4.2

~~REACTIVITY CONTROL SYSTEMS~~

~~3/4.1.2 BORATION SYSTEMS~~

~~3.1.2.1 FLOW PATHS - SHUTDOWN (This specification deleted)~~

~~3.1.2.2 FLOW PATHS - OPERATING (This specification deleted)~~

~~3.1.2.3 CHARGING PUMP - SHUTDOWN (This specification deleted)~~

~~3.1.2.4 CHARGING PUMPS - OPERATING (This specification deleted)~~

~~3.1.2.5 BORATED WATER SOURCES - SHUTDOWN (This specification deleted)~~

~~3.1.2.6 BORATED WATER SOURCES - OPERATING (This specification deleted)~~

~~Pages 3/4 1-8 through 3/4 1-13 are deleted.~~

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~~FIGURE 3.1.2.6 HAS BEEN DELETED~~

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ITS A01 ITS 3.4.2

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.2 3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

Applicability APPLICABILITY: MODES 1 and 2# ~~*~~. A02

ACTION:

ACTION A With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, ~~restore T_{avg} to within its limit within 15 minutes or~~ be in ~~HOT STANDBY~~ MODE 2 with k_{eff} < 1.0 within the next ~~15~~ 30 minutes. A03 A04

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F:

~~a. Within 15 minutes prior to achieving reactor criticality, and~~

~~b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F with the T_{avg}-T_{ref} Deviation Alarm not reset.~~ ~~every 12 hours~~ In accordance with the Surveillance Frequency Control Program L01 LA01

Applicability # With K_{eff} greater than or equal to 1.0.
~~* See Special Test Exception 3.10.3.~~ A02

ITS

A01

ITS 3.4.2

~~REACTIVITY CONTROL SYSTEMS~~

~~3/4.1.2 BORATION SYSTEMS~~

~~3.1.2.1 FLOW PATHS - SHUTDOWN (This specification deleted)~~

~~3.1.2.2 FLOW PATHS - OPERATING (This specification deleted)~~

~~3.1.2.3 CHARGING PUMP - SHUTDOWN (This specification deleted)~~

~~3.1.2.4 CHARGING PUMPS - OPERATING (This specification deleted)~~

~~3.1.2.5 BORATED WATER SOURCES - SHUTDOWN (This specification deleted)~~

~~3.1.2.6 BORATED WATER SOURCES - OPERATING (This specification deleted)~~

~~Pages 3/4 1-8 through 3/4 1-13 are deleted.~~

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~~Figure 3.1.2.6 has been deleted.~~

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DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 The Applicability of CTS 3.1.1.4 is modified by Footnote *, which states "See Special Test Exception 3.10.3." The ITS 3.4.2 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.1.1.4 ACTION states that with a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore T_{avg} to within its limit within 15 minutes or "be in HOT STANDBY within the next 15 minutes." ITS 3.4.2, ACTION A, states that with T_{avg} in one or more RCS loops not within limit, be in MODE 2 with k_{eff} < 1.0 within 30 minutes. This changes the CTS requirement to enter HOT STANDBY to enter MODE 2 with k_{eff} < 1.0. Other changes to this CTS Action are discussed in DOC A04.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.1.1.4 is applicable in MODE 1 and MODE 2 with k_{eff} ≥ 1.0. CTS 3.0.1 states that compliance with the LCO is required during the MODES or other conditions specified therein, except that upon failure to meet the LCO, the associated Action requirements shall be met. Therefore, the CTS 3.1.1.4 ACTION to enter MODE 3 ceases to be applicable once the unit enters MODE 2 with k_{eff} < 1.0, and the Action is exited. As a result, changing the Action to "be in MODE 2 with k_{eff} < 1.0," results in no operational difference from the CTS Action. This change is designated as administrative, as it results in no technical change to the CTS.

A04 CTS 3.1.1.4 ACTION states that with a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, "restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes." ITS 3.4.2, ACTION A, states that with T_{avg} in one or more RCS loops not within limit, be in MODE 2 with k_{eff} < 1.0 within 30 minutes. This changes the CTS by eliminating the redundant and unnecessary requirement to restore T_{avg} to within its limit within 15 minutes. The change associated with entering MODE 2 with k_{eff} < 1.0 instead of HOT STANDBY is discussed in DOC A03.

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DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

This change is acceptable because it results in no technical change to the Technical Specifications. Although CTS 3.1.1.4 ACTION appears to only allow 15 minutes to restore the parameter to within the limit, it actually allows the entire 30 minutes to either restore the parameter or to be in HOT STANDBY (essentially outside the Applicability of CTS 3.1.1.4). In addition, CTS 3.1.1.4 ACTION only requires actual steps to begin reducing reactor power at the beginning of the last 15 minutes of the 30-minute interval. However, CTS 3.0.2 states that if the LCO is restored prior to expiration of the specified interval, completion of the ACTION requirements is not required. Therefore, for this case, if the parameter is restored between 15 minutes and 30 minutes after the LCO is not met, completion of the CTS 3.1.1.4 ACTION to be in HOT STANDBY is not required. Thus, 30 minutes is essentially allowed for either the parameter to be restored to within limit or the unit to be in HOT STANDBY (i.e., only one of the two CTS Actions must be met within 30 minutes). The CTS 3.0.2 requirement is retained in proposed ITS LCO 3.0.2. Therefore, this change does not expand the total time interval allowed to restore the parameter, as a 30-minute interval is already essentially allowed by the CTS. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.1.1.4 requires verification that RCS T_{avg} is greater than or equal to 541°F within 15 minutes prior to achieving reactor criticality, and every 30 minutes when the reactor is critical, RCS T_{avg} is less than 551°F, and the T_{avg} - T_{ref} deviation alarm not reset. ITS SR 3.4.2.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for the SRs and the Bases to the Surveillance Frequency Control Program. (The replacement of the conditional Frequency requirements to verify RCS T_{avg} within limits with a periodic 12 hour Frequency is discussed in DOC L01).

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the

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DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.1.1.4 states that the RCS T_{avg} shall be determined to be ≥ 541°F within 15 minutes prior to achieving reactor criticality, and every 30 minutes when the reactor is critical, RCS T_{avg} is less than 551°F, and the T_{avg} - T_{ref} deviation alarm not reset. ITS SR 3.4.2.1 requires RCS T_{avg} in each loop to be verified ≥ 541°F "In accordance with the Surveillance Frequency Control Program." The specified Surveillance Frequency that is being moved to the Surveillance Frequency Control Program is every 12 hours. This changes the CTS by replacing the requirements for verifying RCS T_{avg} within limits 15 minutes prior to achieving criticality and every 30 minutes when the reactor is critical, RCS T_{avg} is < 551°F, and the T_{avg}-T_{ref} deviation alarm is not reset, with a periodic 12 hour Frequency. Moving the specified Surveillance Frequency to the Surveillance Frequency Control Program is discussed in DOC LA01.

The purpose of CTS 4.1.1.4 is to ensure RCS T_{avg} is within limit when the reactor is critical. The requirement is that RCS T_{avg} be ≥ 541°F when the unit is operating in MODE 1 and MODE 2 with k_{eff} ≥ 1.0. Based on ITS SR 3.0.4, this would require the SR to be met within 12 hours prior to entry into MODE 2 with k_{eff} ≥ 1.0 or before the reactor is critical. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of assurance. The 12 hours Frequency is frequent enough to prevent an inadvertent violation of the LCO. In the approach to criticality, the reactor coolant pumps are adding heat to the RCS, so the conditions before and after criticality are similar. The approach to criticality is a carefully controlled evolution where RCS temperature is closely monitored. Therefore, 12 hours is frequent enough for the Technical Specifications to require recording of T_{avg} prior to criticality, given that it is being carefully monitored. The inoperability of an alarm or with an alarm not reset does not increase the probability of RCS temperature (T_{avg}) being outside its limit. The alarms are for indication only and are not credited in any safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.1.1.4

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ [541]°F. 2

Applicability, Note #

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.1.1.4 SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ [541]°F.	~~[12 hours~~ 2 4 ~~OR~~ In accordance with the Surveillance Frequency Control Program] 4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.1.1.4 LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ [541]°F. 2

Applicability, Note # APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with ~~K~~ k_{eff} < 1.0.	30 minutes 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.1.1.4 SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ [541]°F.	~~[12 hours~~ 2 4 ~~OR~~ In accordance with the Surveillance Frequency Control Program] 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Editorial/grammatical error corrected.

4. ISTS SR 3.4.2.1 (ITS SR 3.4.2.1) provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for this SR under the Surveillance Frequency Control Program.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND	This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical. The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions. The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical. The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen. The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.
APPLICABLE SAFETY ANALYSES	Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical (k_{eff} ≥ 1.0) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with k_{eff} ≥ 1.0, LCO 3.4.2 is applicable since the reactor can only be critical (k_{eff} ≥ 1.0) in these MODES.

~~The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.~~ 2

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with K_{eff} < 1.0 in an orderly manner and without challenging plant systems. 3 3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above ~~[~~541~~]~~°F. ~~[The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.~~ 4 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 6

REFERENCES

1. U FSAR, Section ~~[15.0.3]~~ 15.1. 1 4

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND	This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical. The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions. The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical. The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen. The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.
APPLICABLE SAFETY ANALYSES	Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical (k_{eff} ≥ 1.0) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with k_{eff} ≥ 1.0, LCO 3.4.2 is applicable since the reactor can only be critical (k_{eff} ≥ 1.0) in these MODES.

~~The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.~~ 2

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with K_{eff} < 1.0 in an orderly manner and without challenging plant systems. 3 3

SEQUOYAH UNIT 2 Revision XXX

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above ~~[~~541~~]~~°F. ~~[The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.~~ 4 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 6

REFERENCES

1. U FSAR, Section ~~[15.0.3]~~ 15.1. 1 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2 BASES, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The discussion in the Bases regarding Special Test Exceptions has been deleted since the Special Test Exception LCOs are not normally discussed in the Bases of other LCOs.

3. Editorial/grammatical error corrected.

4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. ISTS SR 3.4.2.1 (ITS SR 3.4.2.1) Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.2.1 under the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.4.3, RCS Pressure and Temperature (P/T) Limits

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

ITS A01 ITS 3.4.3

REACTOR COOLANT SYSTEM

3/4.4.9 RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.9.1 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

Applicability APPLICABILITY: At all times.

ACTIONS:

ACTION A a. With the requirements of the LCO not met in MODE 1, 2, 3, or 4, restore the parameter(s) to within limits in 30 minutes and determine RCS is acceptable* for continued operation within 72 hours.
ACTION B With the required action above not met, be in MODE 3 within the next 6 hours and in MODE 5, with RCS pressure < 500 psig, within the following 30 hours.

ACTION C b. With the requirements of the LCO not met any time other than MODE 1, 2, 3, or 4, immediately initiate action to restore parameter(s) to within limits and, prior to entering MODE 4, determine RCS is acceptable* for continued operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 4.4.9.1.1 Verify** RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR ~~every 30 minutes~~.

In accordance with the Surveillance Frequency Control Program LA01

ACTION A Note, ACTION C Note * The determination that the RCS is acceptable for continued operation must be completed for any entry into Action (a) or (b).

SR 3.4.3.1 Note ** Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing.

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FIGURE 3.4-2 - DELETED

SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMINATIONS
APPLICABLE UP TO 16 EFPY

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ITS

A01

ITS 3.4.3

FIGURE 3.4-3 - DELETED

SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 16 EFPY

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REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 DELETED.

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REACTOR COOLANT SYSTEM

3/4.4.10 DELETED

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REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM HEAD VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 This specification has been deleted.

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ITS A01 ITS 3.4.3

REACTOR COOLANT SYSTEM

3/4.4.9 RCS PRESSURE/TEMPERATURE (P/T) LIMITS

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.9.1 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

Applicability APPLICABILITY: At all times.

ACTIONS:

ACTION A a. With the requirements of the LCO not met in MODE 1, 2, 3, or 4, restore the parameter(s) to within limits in 30 minutes and determine RCS is acceptable* for continued operation within 72 hours.

ACTION B With the required action above not met, be in MODE 3 within the next 6 hours and in MODE 5, with RCS pressure < 500 psig, within the following 30 hours.

ACTION C b. With the requirements of the LCO not met any time other than MODE 1, 2, 3, or 4, immediately initiate action to restore parameter(s) to within limits and, prior to entering MODE 4, determine RCS is acceptable* for continued operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 4.4.9.1.1 Verify** RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR ~~every 30 minutes~~ In accordance with the Surveillance Frequency Control Program. LA01

ACTION A Note, ACTION C Note * The determination that the RCS is acceptable for continued operation must be completed for any entry into Action (a) or (b).

SR 3.4.3.1 Note ** Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing.

FIGURE 3.4-2 - DELETED

SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMINATIONS
APPLICABLE UP TO 32 EFPY

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FIGURE 3.4-3 - DELETED

SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 32 EFPY

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REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 DELETED.

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REACTOR COOLANT SYSTEM

3/4.4.10 DELETED

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REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM HEAD VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 This specification has been deleted.

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DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.9.1.1 requires verification that RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR every 30 minutes during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ITS SR 3.4.3.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for the SR and the Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

3.4.9.1 LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

Applicability APPLICABILITY: At all times.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION Note * / ACTION a	A. ------------NOTE------------ Required Action A.2 shall be completed whenever this Condition is entered. --------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
ACTION a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < ~~[~~500~~]~~ psig.	6 hours 36 hours
ACTION Note * / ACTION b	C. ------------NOTE------------ Required Action C.2 shall be completed whenever this Condition is entered. --------------------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

2

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.9.1.1 Note ** SR 3.4.3.1	-------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---		
4.4.9.1.1	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

3.4.9.1 LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

Applicability APPLICABILITY: At all times.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION Note *	A. ------------NOTE------------ Required Action A.2 shall be completed whenever this Condition is entered. --------------------------------	A.1 Restore parameter(s) to within limits.	30 minutes
		AND	
ACTION a	Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.2 Determine RCS is acceptable for continued operation.	72 hours
ACTION a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
		AND	
		B.2 Be in MODE 5 with RCS pressure < ~~[~~500~~]~~ psig.	36 hours
ACTION Note *	C. ------------NOTE------------ Required Action C.2 shall be completed whenever this Condition is entered. --------------------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
		AND	
ACTION b	Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.4.9.1.1 Note ** SR 3.4.3.1	------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
4.4.9.1.1	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS SR 3.4.3.1 (ITS SR 3.4.3.1) provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for this SR under the Surveillance Frequency Control Program.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and

BASES

APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed ~~before continuing operation~~ within 72 hours. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. 2

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < ~~[~~500~~]~~ psig within 36 hours. 3

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

BASES

ACTIONS (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required when RCS pressure and temperature conditions are undergoing planned changes. ~~[This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-7924-A, April 1975.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. Regulatory Guide 1.99, Revision 2, May 1988.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and

BASES

APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed ~~before continuing operation~~ within 72 hours. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. 2

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < ~~[~~500~~]~~ psig within 36 hours. 3

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

BASES

ACTIONS (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required when RCS pressure and temperature conditions are undergoing planned changes. ~~[This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-7924-A, April 1975.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. Regulatory Guide 1.99, Revision 2, May 1988.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3 BASES, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes are made to reflect the Specification.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.4.3.1 Bases provide two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for ITS SR 3.4.3.1 under the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 4

ITS 3.4.4, RCS Loops – MODES 1 and 2

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

ITS A01 ITS 3.4.4

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

OPERABLE and A02

LCO 3.4.4 3.4.1.1 All reactor coolant loops shall be in operation.

Applicability APPLICABILITY: MODES 1 and 2.~~*~~ A03

ACTION:

ACTION With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within ~~1~~ 6 hour. L01

SURVEILLANCE REQUIREMENT

SR 3.4.4.1 4.4.1.1 The above required reactor coolant loops shall be verified to be in operation ~~and circulating reactor coolant at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program. LA01 LA02 LA01

~~*See Special Test Exception 3.10.4.~~ A03

ITS

A01

ITS 3.4.4

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

OPERABLE and — A02

LCO 3.4.4 3.4.1.1 All reactor coolant loops shall be in operation.

Applicability APPLICABILITY: MODES 1 and 2.~~*~~ — A03

ACTION:

ACTION With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within ~~1~~ 6 hour. — L01

SURVEILLANCE REQUIREMENT

SR 3.4.4.1 4.4.1.1 The above required reactor coolant loops shall be verified to be in operation ~~and circulating reactor coolant~~ ~~at least once per 12 hours~~. — LA01

In accordance with the Surveillance Frequency Control Program — LA02

LA01

~~*See Special Test Exception 3.10.4.~~ — A03

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DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.1.1 states that all reactor coolant loops shall be in operation. ITS 3.4.4 states that four RCS loops shall be OPERABLE and in operation. This changes the CTS by requiring the RCS loops to be OPERABLE.

This change is acceptable because it is consistent with the current use and understanding of the LCO. It is not sufficient for an RCS loop to be in operation if it is not capable of performing its safety function (i.e., OPERABLE). This change is designated as administrative as it clarifies the current understanding of a requirement.

A03 The Applicability of CTS 3.4.1.1 is modified by footnote * that states "See Special Test Exception 3.10.4." The ITS 3.4.4 Applicability does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS to not include the footnote "See Special Test Exception 3.10.4."

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.1 states that the required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.4.1 states that each RCS loop shall be verified to

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DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

be in operation in accordance with the Surveillance Frequency Control Program. This changes the CTS by moving the Surveillance Requirement to verify that the reactor coolant loops are circulating reactor coolant to the Bases. (See DOC LA02 for the discussion for moving the Surveillance Frequency to the Surveillance Frequency Control Program.)

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a RCS loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.1.1 requires the RCS loops to be verified in operation at least once per 12 hours. ITS SR 3.4.4.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for the SR and the Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.1.1 ACTION states when the reactor coolant loop requirements are not met, the unit must be in HOT

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DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

STANDBY within 1 hour. ITS 3.4.4 ACTION A states when the RCS loop requirements are not met, the unit must be in MODE 3 within 6 hours. This changes the CTS by relaxing the Completion Time from 1 hour to 6 hours.

The purpose of CTS 3.4.1.1 ACTION is to require a unit shutdown if the necessary reactor coolant loop flow is not available. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capability and capacity of the remaining systems or features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. Operating experience has shown that 6 hours is a reasonable time to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. It is likely that failure to meet the LCO requirements will lead to a reactor trip on low flow. However, if the LCO is not met for a reason that does not lead to a reactor trip, then 6 hours to transition from full power operation to MODE 3 is consistent with the Completion Time provided for a loss of safety function for other systems and with LCO 3.0.3. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

3.4.1.1 LCO 3.4.4 ~~[~~Four~~]~~ RCS loops shall be OPERABLE and in operation. 1

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION	A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.1	SR 3.4.4.1	Verify each RCS loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

3.4.1.1 LCO 3.4.4 ~~[~~Four~~]~~ RCS loops shall be OPERABLE and in operation. 1

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION	A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.1	SR 3.4.4.1	Verify each RCS loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS SR 3.4.4.1 (ITS SR 3.4.4.1) provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for this SR under the Surveillance Frequency Control Program.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission,

b. Improving the neutron economy by acting as a reflector,

c. Carrying the soluble neutron poison, boric acid,

d. Providing a second barrier against fission product release to the environment, and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through ~~[~~four~~]~~ loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control. 1

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming ~~[~~four~~]~~ RCS loops are in operation. The majority of the plant 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the INSERT 1 ~~[four] pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events~~ (Ref. 1). 2

Steady state DNB analysis has been performed for the ~~[~~four~~]~~ RCS loop operation. INSERT 2 ~~For [four] RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.~~ 1 2

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, ~~[~~four~~]~~ pumps are required at rated power. 1

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an associated OPERABLE SG. 2

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

2 **INSERT 1**

partial and complete loss of reactor coolant flow, rod withdrawal events, startup of an inactive reactor coolant loop, and single RCP locked rotor

2 **INSERT 2**

These analyses establish typical allowable RCS loop average temperature and ΔT for the design power distribution and flow as a function of RCS pressure. These analyses also establish a locus of power, pressure, and temperature conditions for which the departure from nucleate boiling ratio (DNBR) is equal to its Safety Limit value. The area of permissible operation is bounded by the combination of assumed reactor trips for high neutron flux (fixed setpoint), high pressure (fixed setpoint), low pressure (fixed setpoint), overtemperature ΔT (variable setpoint), and overpower ΔT (variable setpoint). The difference between the reactor trip values assumed in the safety analyses and the nominal reactor trip setpoints provides an allowance for instrumentation channel error and setpoint error.

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and 3
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 3

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

REFERENCES

1. U FSAR, Chapter ~~Section~~ 15 ~~[ ]~~. 2 1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission,

b. Improving the neutron economy by acting as a reflector,

c. Carrying the soluble neutron poison, boric acid,

d. Providing a second barrier against fission product release to the environment, and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through ~~[~~four~~]~~ loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control. 1

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming ~~[~~four~~]~~ RCS loops are in operation. The majority of the plant 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the INSERT 1 ~~[four] pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events~~ (Ref. 1). 2

Steady state DNB analysis has been performed for the ~~[~~four~~]~~ RCS loop operation. INSERT 2 ~~For [four] RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.~~ 1 2

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, ~~[~~four~~]~~ pumps are required at rated power. 1

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an associated OPERABLE SG. 2

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

2 **INSERT 1**

partial and complete loss of reactor coolant flow, rod withdrawal events, startup of an inactive reactor coolant loop, and single RCP locked rotor

2 **INSERT 2**

These analyses establish typical allowable RCS loop average temperature and ΔT for the design power distribution and flow as a function of RCS pressure. These analyses also establish a locus of power, pressure, and temperature conditions for which the departure from nucleate boiling ratio (DNBR) is equal to its Safety Limit value. The area of permissible operation is bounded by the combination of assumed reactor trips for high neutron flux (fixed setpoint), high pressure (fixed setpoint), low pressure (fixed setpoint), overtemperature ΔT (variable setpoint), and overpower ΔT (variable setpoint). The difference between the reactor trip values assumed in the safety analyses and the nominal reactor trip setpoints provides an allowance for instrumentation channel error and setpoint error.

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and 3
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 3

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 5

REFERENCES

1. U FSAR, ~~Section~~ Chapter ~~[ ]~~ 15. 2 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes are made to reflect the Specification.

4. ISTS SR 3.4.4.1 Bases provide two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for ITS SR 3.4.4.1 under the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 5

ITS 3.4.5, RCS Loops – MODE 3

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

ITS A01 ITS 3.4.5

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.5 3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with at least two reactor coolant loops in operation when ~~the Reactor Trip System breakers are closed~~ the Rod Control System is capable of rod withdrawal and at least one reactor coolant loop in operation when the ~~Reactor Trip System breakers are open~~ the Rod Control System is not capable of rod withdrawal:* LA01

~~a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,~~

~~b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,~~

~~c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,~~

~~d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,~~

LA02

Applicability APPLICABILITY: MODE 3

ACTION:

ACTION A, ACTION B a. With one less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. M01

ACTION C b. With only one reactor coolant loop in operation and ~~the Reactor Trip System breakers in the closed position~~ the Rod Control System capable of rod withdrawal, within 1 hour ~~open the Reactor Trip System breakers~~ place Rod Control System in condition incapable of rod withdrawal. LA01 LA01

ACTION D c. With no reactor coolant loop in operation or two required RCS loops inoperable, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return the required coolant loop to OPERABLE status and operation. M01 M01

Add proposed Required Action D.1 M01

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.5.3 Note L01

SR 3.4.5.3 4.4.1.2.1 At least the above required reactor coolant pumps, ~~if not in operation~~, shall be determined to be OPERABLE ~~once per 7 days~~ In accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability. LA03

SR 3.4.5.2 4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 21 percent ~~at least once per 12 hours~~.

SR 3.4.5.1 4.4.1.2.3 The required Reactor Coolant loops shall be verified to be in operation ~~and circulating reactor coolant at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program. LA04 LA03

LCO Note * * All reactor coolant pumps may be de-energized for up to 1 hour per 8 hour period provided (1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature. LA04 M02

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ITS A01 ITS 3.4.5

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

LA01

LCO 3.4.5 3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with at least two reactor coolant loops in operation when ~~the Reactor Trip System breakers are closed~~ the Rod Control System is capable of rod withdrawal and at least one reactor coolant loop in operation when ~~the Reactor Trip System breakers are open~~ the Rod Control System is not capable of rod withdrawal:*

LA02

~~a. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant pump,~~

~~b. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant pump,~~

~~c. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant pump,~~

~~d. Reactor Coolant Loop D and its associated steam generator and Reactor Coolant pump.~~

Applicability APPLICABILITY: MODE 3

ACTION:

M01

ACTION A / ACTION B a. With one less than the above required Reactor Coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

LA01

ACTION C b. With only one reactor coolant loop in operation and ~~the Reactor Trip System breakers in the closed position~~ the Rod Control System capable of rod withdrawal, within one hour ~~open the Reactor Trip System breakers~~ place Rod Control System in condition incapable of rod withdrawal.

LA01

ACTION D c. With no Reactor Coolant loop in operation or two required RCS loops inoperable, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return the required Reactor Coolant loop to OPERABLE status and operation.

M01

Add proposed Required Action D.1

M01

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.5.3 Note

L01

SR 3.4.5.3 4.4.1.2.1 At least the above required Reactor Coolant pumps, ~~if not in operation~~, shall be determined to be OPERABLE ~~once per 7 days~~ In accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

LA03

SR 3.4.5.2 4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 21 percent ~~at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program.

LA04

SR 3.4.5.1 4.4.1.2.3 The required Reactor Coolant loops shall be verified to be in operation ~~and circulating reactor coolant at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program.

LA03

LA04

LCO Note * * All Reactor Coolant pumps may be de-energized for up to 1 hour per 8 hour period provided (1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

M02

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS – MODE 3

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.2 ACTION a states that with less than the required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours. CTS 3.4.1.2 ACTION b states that, with only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, open the Reactor Trip System breakers within 1 hour. CTS 3.4.1.2 ACTION c states that, with no reactor coolant loops in operation, suspend operations that would involve a reduction in boron concentration of the RCS and initiate action to return the required coolant loop to operation. ITS 3.4.5 ACTION A specifies the Required Action for one required RCS loop inoperable. The Required Action is to restore the RCS loop to OPERABLE status within 72 hours. ITS 3.4.5 ACTION C specifies the Required Action for one required RCS loop not in operation with the Rod Control System capable of rod withdrawal. The Required Action is to place the Rod Control System in a condition incapable of rod withdrawal within 1 hour. ITS 3.4.5 ACTION D specifies the Required Actions for two required RCS loops inoperable and for no required RCS loop in operation (i.e., two required RCS loops not in operation with Rod Control System capable of rod withdrawal or the required RCS loop not in operation with Rod Control System not capable of rod withdrawal). The Required Actions are to immediately place the Rod Control System in a condition incapable of rod withdrawal, immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, and to immediately initiate action to restore one RCS loop to OPERABLE status and operation. This changes the CTS by revising the Actions to immediately require actions to be taken when two required RCS loops are inoperable or two RCS loops are not in operation when the Rod Control System is capable of rod withdrawal.

This change is acceptable because it provides appropriate actions for a loss of two required RCS loops. Under these conditions, immediate action is necessary to ensure certain unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. This change is designated as more restrictive, because it requires immediate action instead of allowing time for restoration.

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS – MODE 3

M02 CTS 3.4.1.2 states that at least two reactor coolant loops shall be OPERABLE with at least two reactor coolant pumps in operation when the Reactor Trip System breakers are closed and at least one reactor coolant pump in operation when the Reactor Trip System breakers are open. This requirement is modified by footnote * that states all reactor coolant pumps may be de-energized for up to 1 hour under the conditions specified therein. ITS 3.4.5 contains the same allowance, but limits the use of the 1 hour exception to once per 8 hour period. This changes the CTS by modifying the 1 hour allowance that all reactor coolant pumps may be de-energized and limits the usage of the allowance to once per 8 hour period.

The purpose of the 1 hour allowance is to allow a reactor coolant loop to be removed from operation to support testing. This change is acceptable because it helps ensure that boron stratification and inadequate decay heat removal do not occur should multiple 1 hour periods be required. This change is designated as more restrictive because it limits the allowance to 1 hour per 8 hour period, and that restriction does not currently exist.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.1.2 specifies requirements for RCS loops when the Reactor Trip System breakers are in the closed position, and requirements for RCS loops when the Reactor Trip System breakers are in the open position. With only one RCS loop in operation and the Reactor Trip System breakers in the closed position, CTS 3.4.1.2, ACTION b requires the Reactor Trip System breakers to be opened within 1 hour. ITS LCO 3.4.5.a specifies requirements for the RCS loops when the Rod Control System is capable of rod withdrawal. ITS LCO 3.4.5.b specifies requirements for the RCS loops when the Rod Control System is not capable of rod withdrawal. ITS 3.4.5 ACTION C requires the Rod Control System to be placed in a condition incapable of rod withdrawal when one required RCS loop is not in operation with the Rod Control System capable of rod withdrawal. This changes the CTS by moving the details on how to place the Rod Control System in a state capable or incapable of rod withdrawal (i.e., by using the Reactor Trip System breakers) from the Technical Specifications to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still continues to specify requirements on the RCS depending on the status of the Rod Control System's capability to withdraw rods. These procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS – MODE 3

changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2 contains a description of what constitutes an OPERABLE RCS loop. ITS 3.4.5 does not include a description of what constitutes an OPERABLE RCS loop. This changes the CTS by moving the details of what constitutes an OPERABLE RCS loop to the Bases.

The removal of these details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for the RCS loops to be OPERABLE. The removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LA03 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.1.2.1 requires verification of the correct breaker alignments and indicated power availability for the required reactor coolant pumps once per 7 days. ITS SR 3.4.5.3 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." CTS 4.4.1.2.2 requires verification that the steam generator secondary water level is within limits at least once per 12 hours. ITS SR 3.4.5.2 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." CTS 4.4.1.2.3 requires verification that the required RCS loops are in operation at least once per 12 hours. ITS SR 3.4.5.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for the SR and the Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS – MODE 3

as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA04 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.3 states that the required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.5.1 states that the required reactor coolant loops shall be verified to be in operation in accordance with the Surveillance Frequency Control Program. (See DOC LA03 for a discussion of moving the "12 hour" Surveillance Frequency to the Surveillance Frequency Control Program.) This changes the CTS by moving the requirement to verify that the reactor coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a reactor coolant loop be in operation, and a loop that is in operation will be circulating reactor coolant. As described in the ITS Bases, verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring. These types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.4.1.2.1 states that the required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE by verifying correct breaker alignment and indicated power availability. ITS SR 3.4.5.3 requires verification of correct breaker alignment and indicated power availability to each required pump. It is modified by a Note that states "Not required to be performed until 24 hours after a required pump is not in operation." This changes the CTS by not requiring the SR to be performed until 24 hours after a pump is taken out of operation.

The purpose of CTS 4.4.1.2.1 is to ensure that the standby reactor coolant pump is ready to operate. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Note provides time to perform the Surveillance to verify correct breaker alignment and indicated power availability. Without the Note, the Surveillance would not be met immediately after taking a pump out of operation. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

3.4.1.2 LCO 3.4.5 ~~[~~Two~~]~~ RCS loops shall be OPERABLE and either: 1

a. ~~[~~Two~~]~~ RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal ; or 1 2

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

3.4.1.2 Note *

---NOTE---

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM "SHUTDOWN MARGIN (SDM)" of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

Applicability APPLICABILITY: MODE 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours	
ACTION a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hour s	2

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION b	C. [One required RCS loop not in operation with Rod Control System capable of rod withdrawal.	~~C.1~~	~~Restore required RCS loop to operation.~~	~~1 hour~~	1
		~~OR~~			4
		C.~~2~~ 1	Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour]	1
ACTION c	D. [Two] [required] RCS loops inoperable.	D.1	Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately	1
	OR	AND			
DOC M01	Required RCS loop(s) not in operation.	D.2	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately	
		AND			
		D.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately	

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SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.1.2.3	SR 3.4.5.1	Verify required RCS loops are in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 5
4.4.1.2.2	SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ [17]% for required RCS loops. 21	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 1 5
4.4.1.2.1 DOC L01	SR 3.4.5.3	------------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 5

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

3.4.1.2 LCO 3.4.5 [Two] RCS loops shall be OPERABLE and either: 1

a. [Two] RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal ; or 1 2

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

3.4.1.2 Note *

--NOTE--
All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the ~~SDM~~ "SHUTDOWN MARGIN (SDM)" of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

Applicability APPLICABILITY: MODE 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours	
ACTION a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hour s	2

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION b	C. [One required RCS loop not in operation with Rod Control System capable of rod withdrawal.	~~C.1~~	~~Restore required RCS loop to operation.~~	~~1 hour~~	1
		~~OR~~			4
		C.~~2~~ 1	Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour]	1
ACTION c	D. [Two] [required] RCS loops inoperable.	D.1	Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately	1
	OR	AND			
DOC M01	Required RCS loop(s) not in operation.	D.2	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately	
		AND			
		D.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately	

SURVEILLANCE REQUIREMENTS

		SURVEILLANCE	FREQUENCY	
4.4.1.2.3	SR 3.4.5.1	Verify required RCS loops are in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 5
4.4.1.2.2	SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ ~~[17]~~ 21 % for required RCS loops.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 1 5
4.4.1.2.1 DOC L01	SR 3.4.5.3	------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power are available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 5

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5, RCS LOOPS – MODE 3

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Typographical error corrected and editorial change made for enhanced clarity.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The Required Actions of ISTS 3.4.5 Condition C require restoration of the required RCS loop to operation, or the placement of the Rod Control System in a condition incapable of rod withdrawal. The Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 4.1.6.g, states "A Required Action which requires restoration, such that the Condition is no longer met, is considered superfluous. It is only included if it would be the only Required Action for the Condition or it is needed for presentation clarity." Neither exception applies in this case. Therefore, Required Action C.1 is deleted and the subsequent Required Action is renumbered.

5. ISTS SR 3.4.5.1 (ITS SR 3.4.5.1), ISTS SR 3.4.5.2 (ITS SR 3.4.5.2), and ISTS SR 3.4.5.3 (ITS SR 3.4.5.3) provide two options for controlling the Frequencies of the Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND	In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. The reactor coolant is circulated through ~~[~~four~~]~~ RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. 1 In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, ~~[~~two~~]~~ RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal. 1
APPLICABLE SAFETY ANALYSES	Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM. Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least ~~[~~two~~]~~ RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation ~~to be consistent with MODE 3 accident analyses~~ to ensure removal of decay heat from the core and a homogeneous boron concentration throughout the RCS. 1 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least ~~[~~two~~]~~ RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, ~~[~~two~~]~~ RCS loops must be in operation. ~~[~~Two~~]~~ RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents. 1 1

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one associated OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required. 3

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~;"
LCO 3.4.6, "RCS Loops - MODE 4~~,~~;"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled~~,~~;" 4
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),~~; and 5 4
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 5

BASES

ACTIONS

A.1

If one ~~[~~required~~]~~ RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period. 1

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

~~[~~ C.1 ~~and C.2~~ 1 6

If one required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is ~~either to restore the required RCS loop to operation or~~ to place the Rod Control System in a condition incapable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Time~~s~~ of 1 hour~~,~~ to ~~restore the required RCS loop to operation or~~ defeat the Rod Control System is adequate to perform this ~~these~~ operation~~s~~ in an orderly manner without exposing the unit to risk for an undue time period. ~~]~~ 6 1

D.1, D.2, and D.3

If ~~[~~two~~]~~ ~~[~~required~~]~~ RCS loops are inoperable or a required RCS loop is not in operation, except as during conditions permitted by the Note in the LCO section, the Rod Control System must be placed in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving 1

BASES

ACTIONS (continued)

introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 8

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ ~~[17]~~ 21 % for required RCS loops. If the SG secondary side narrow range water level is < ~~[17]~~ 21 %, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal 1 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

of the decay heat. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 8

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 8

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

BASES

REFERENCES	None.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through ~~[~~four~~]~~ RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. 1

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, ~~[~~two~~]~~ RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal. 1

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least ~~[~~two~~]~~ RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation ~~to be consistent with MODE 3 accident analyses~~. 1 2

to ensure removal of decay heat from the core and a homogeneous boron concentration throughout the RCS.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least ~~[~~two~~]~~ RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, ~~[~~two~~]~~ RCS loops must be in operation. ~~[~~Two~~]~~ RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents. 1 1

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one associated OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required. 3

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~;"
LCO 3.4.6, "RCS Loops - MODE 4~~,~~;"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled~~,~~;" 4
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),~~; and 5 4
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 5

BASES

ACTIONS

A.1

If one ~~[~~required~~]~~ RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period. 1

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

~~[~~ C.1 ~~and C.2~~ 1 6

If one required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is ~~either to restore the required RCS loop to operation or~~ to place the Rod Control System in a condition incapable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Time~~s~~ of 1 hour~~,~~ to ~~restore the required RCS loop to operation or~~ defeat the Rod Control System is adequate to perform this ~~these~~ operation~~s~~ in an orderly manner without exposing the unit to risk for an undue time period. ~~]~~ 6 1

D.1, D.2, and D.3

If ~~[~~two~~]~~ ~~[~~required~~]~~ RCS loops are inoperable or a required RCS loop is not in operation, except as during conditions permitted by the Note in the LCO section, the Rod Control System must be placed in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving 1

BASES

ACTIONS (continued)

introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 8

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ ~~[17]~~ 21 % for required RCS loops. If the SG secondary side narrow range water level is < ~~[17]~~ 21 %, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal 1 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

of the decay heat. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 8

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 7

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 8

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

BASES

REFERENCES	None.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5 Bases, RCS LOOPS – MODE 3

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes have been made to be consistent with other statements in the Bases.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. Editorial/grammatical errors have been corrected.

5. Changes are made to reflect the Specification.

6. Changes are made to reflect changes made to the Specification.

7. ISTS SR 3.4.5.1 through SR 3.4.5.3 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.5.1 through SR 3.4.5.3 under the Surveillance Frequency Control Program.

8. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.5, RCS LOOPS – MODE 3

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 6

ITS 3.4.6, RCS LOOPS – MODE 4

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

ITS A01 ITS 3.4.6

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.4.6 3.4.1.3 a. At least two of the reactor coolant and/or residual heat removal (RHR) loops ~~listed below~~ shall be OPERABLE:

~~1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,~~*

~~2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,~~*

~~3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,~~*

~~4. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,~~*

~~5. Residual Heat Removal Loop A,~~

~~6. Residual Heat Removal Loop B.~~

LA01

LCO 3.4.6 b. At least one of the ~~above~~ reactor coolant and/or RHR loops shall be in operation.**

Applicability APPLICABILITY: MODE 4.

M01

ACTION:

one

Add proposed Required Action A.2 Note.

ACTION A a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible, be in COLD SHUTDOWN within ~~20~~ 24 hours.

L01

Add proposed Condition B, first part

M01

ACTION B b. With no reactor coolant or RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return ~~the required~~ one coolant loop to OPERABLE status and operation.

M02

M01

per 8 hour period

removed from operation

A02

LCO 3.4.6 Note 1 **All reactor coolant pumps and residual heat removal pumps may be ~~de-energized~~ removed from operation for up to 1 hour per 8 hour period provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.6 Note 2 *A reactor coolant pump shall not be restarted unless a steam bubble exists in the pressurizer.

ITS

A01

ITS 3.4.6

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

M03

or RHR

Not required to be performed until 24 hours after a required pump is not in operation.

L02

SR 3.4.6.3 4.4.1.3.1 The required reactor coolant pump(s), ~~if not in operation~~, shall be determined to be OPERABLE ~~once per 7 days~~ by verifying correct breaker alignments and indicated power availability.

In accordance with the Surveillance Frequency Control Program

LA02

21% ~~(narrow range indication)~~

M04

SR 3.4.6.2 4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~10 percent (wide-range indication) at least once per 12 hours~~.

LA03

In accordance with the Surveillance Frequency Control Program

LA02

SR 3.4.6.1 4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified to be in operation ~~and circulating reactor coolant at least once per 12 hours~~.

LA04

In accordance with the Surveillance Frequency Control Program

LA02

ITS A01 ITS 3.4.6

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.4.6 3.4.1.3 a. At least two of the reactor coolant and/or Residual heat removal (RHR) loops ~~listed below~~ shall be OPERABLE:

~~1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,~~*

~~2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,~~*

~~3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,~~* LA01

~~4. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,~~*

~~5. Residual Heat Removal Loop A,~~

~~6. Residual Heat Removal Loop B.~~

LCO 3.4.6 b. At least one of the ~~above~~ reactor coolant and/or RHR loops shall be in operation.**

M01

Applicability APPLICABILITY: MODE 4.

ACTION:

one

Add proposed Required Action A.2 Note.

ACTION A a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible, be in COLD SHUTDOWN within ~~20~~ 24 hours. L01

Add proposed Condition B, first part

ACTION B b. With no reactor coolant or RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return ~~the required~~ coolant loop to OPERABLE status and operation. M01

M01 one

removed from operation A02

LCO 3.4.6 Note 1 **All reactor coolant pumps and residual heat removal pumps may be ~~de-energized~~ removed from operation for up to 1 hour per 8 hour period provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1, and 2) core outlet temperature is maintained at least 10°F below saturation temperature. M02

LCO 3.4.6 Note 2 *A reactor coolant pump shall not be restarted unless a steam bubble exists in the pressurizer.

ITS

A01

ITS 3.4.6

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

M03

or RHR

Not required to be performed until 24 hours after a required pump is not in operation.

L02

SR 3.4.6.3 4.4.1.3.1 The required reactor coolant pump(s), ~~if not in operation~~, shall be determined to be OPERABLE ~~once per 7 days~~ by verifying correct breaker alignments and indicated power availability.

In accordance with the Surveillance Frequency Control Program

LA02

SR 3.4.6.2 4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~10 percent (wide-range indication) at least once per 12 hours~~.

LA03

21% ~~(narrow range indication)~~

M04

SR 3.4.6.1 4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified to be in operation ~~and circulating reactor coolant at least once per 12 hours~~.

LA04

In accordance with the Surveillance Frequency Control Program

LA02

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DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.1.3 Footnote ** allows all reactor coolant pumps and residual heat removal (RHR) pumps to be de-energized. ITS LCO 3.4.6 Note 1 allows all reactor coolant pumps and RHR pumps to be removed from operation. This changes the CTS by replacing the word "de-energized" with "removed from operation."

The purpose of CTS 3.4.1.3 Footnote ** is to allow the pumps to not meet the requirement of CTS 3.4.1.3 to be in operation. The change better reflects the deviation to the LCO. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.3 ACTION a states, in part, that with less than the two required loops OPERABLE, immediate action must be taken to return the required loops to OPERABLE status or the unit shall be placed in cold shutdown. CTS 3.4.1.3 ACTION b states, in part, that with no reactor coolant or RHR loop in operation, all operations involving a reduction in boron concentration of the RCS must be suspended and action must be initiated to return the required coolant loop to operation. ITS 3.4.6, ACTION A applies when one of the required loops is inoperable and requires immediate action be initiated to restore the inoperable loop to an OPERABLE status or place the unit in MODE 5. ITS 3.4.6, ACTION B, applies when two of the required loops are inoperable or the required loop is not in operation. In addition to requiring an inoperable loop be restored to OPERABLE status, ITS 3.4.6 ACTION B specifies required actions to prevent boron dilution. This changes the CTS by imposing additional immediate actions to take when both required loops are inoperable.

The purpose of the CTS 3.4.1.3 ACTIONS is to provide appropriate compensatory actions for the condition of one or both required loops inoperable and the condition of no loop in operation. The proposed change requires additional actions be taken that are not required in the CTS for two inoperable loops. This change is acceptable because it will require immediate actions to preclude boron dilution when no reactor coolant or RHR loops are OPERABLE. The additional ITS 3.4.6 ACTION to preclude dilution of the RCS boron concentration provides increased assurance of safe plant operation. The proposed change is designated as more restrictive because it requires additional

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DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

immediate actions to preclude RCS boron concentration reductions when no OPERABLE reactor coolant or RHR loops are available for forced circulation of the RCS.

M02 CTS LCO 3.4.1.3.b states, in part, that at least two coolant loops shall be OPERABLE and at least one must be in operation. This requirement is modified by a Footnote** that states, in part, that all reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour. ITS 3.4.6 contains the same allowance, but limits the use of the 1 hour exception to once per 8 hour period. This changes the CTS to limit the use of the 1 hour exception to once per 8 hour period.

The purpose of the 1 hour allowance is to allow a coolant loop to be removed from operation to support testing. This change is acceptable because it helps ensure that boron stratification and inadequate decay heat removal do not occur should multiple 1 hour periods be required. This change is designated as more restrictive because it limits an allowance to 1 hour per 8 hour period, and that restriction does not currently exist.

M03 CTS 4.4.1.3.1 states, in part, the required reactor coolant pump(s), if not in operation, shall be determined OPERABLE by verifying correct breaker alignment and indicated power availability. ITS SR 3.4.6.3 requires verification that correct breaker alignment and indicated power are available to the required pump not in operation. ITS LCO 3.4.6 allows a combination of reactor coolant pumps and RHR pumps. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required RHR pumps that are not in operation.

The purpose of the CTS is to ensure a standby pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional reactor coolant pump or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it requires performance of Surveillance on RHR pumps in addition to reactor coolant pumps.

M04 CTS 3.4.1.3.a requires two reactor coolant and/or residual heat removal (RHR) loops to be OPERABLE in MODE 4. Each required reactor coolant loop is required to be accompanied by its associated steam generator and reactor coolant pump. CTS 4.4.1.3.2 requires the OPERABILITY of required steam generators to be determined by verifying the secondary side water level to be greater than or equal to 10 percent (wide range indication). ITS SR 3.4.6.2 requires verification that each required steam generator has a secondary side water level ≥ 21% (narrow range indication). (See DOC LA03 for the discussion of moving the procedural details of the specific indicator to be used to verify the secondary side steam generator water level to the Bases.) This changes the CTS by requiring a higher secondary side steam generator water level for determining steam generator OPERABILITY.

The purpose of CTS 3.4.1.3 is to ensure adequate means of decay heat removal and boron mixing is OPERABLE in MODE 4. When a reactor coolant loop is

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used to meet the requirement, the associated steam generator OPERABILITY is required to be determined by verifying the secondary side water level is high enough to ensure the steam generator tubes are not uncovered, thereby providing the heat sink necessary for decay heat removal. A secondary side steam generator water level of 21% (narrow range indicator) will ensure that at least 76 inches of water is above the top of the steam generators tubes in MODES 1, 2, 3, 4 and 5. This change is acceptable because requiring a secondary side steam generator water level of 21% (narrow range indicator) will ensure the steam generator tubes remain covered over the entire temperature range of MODE 4. This change is designated as more restrictive because it requires a higher secondary side steam generator water level for steam generator OPERABILITY than was required in CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.3.a provides the requirements for reactor coolant and/or RHR loops in MODE 4 and includes a description of what constitutes an OPERABLE loop. ITS 3.4.6 provides the requirements for reactor coolant and/or RHR loops in MODE 4, but does not include a description of what constitutes an OPERABLE loop. This changes the CTS by moving the details of what constitutes an OPERABLE coolant loop to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the RCS loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.1.3.1 requires the reactor coolant pump(s), if not in operation, to be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability. CTS 4.4.1.3.2 requires the required steam generator(s) to be demonstrated OPERABLE by verifying secondary side water level to be greater than or equal to 10 percent (wide range indicator) at least once per 12 hours. CTS 4.4.1.3.3 requires one reactor coolant pump or RHR loop to be verified in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.6.1, SR 3.4.6.2, and SR 3.4.6.3 require similar Surveillances, but specify the periodic Frequency as "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

specified Frequencies for the SRs to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.3.2 requires, in part, verifying that secondary side water level in each required steam generator to be greater than or equal to a specified value on the wide range indicator. ITS SR 3.4.6.2 requires verification that the secondary side water level in each required steam generator is greater than or equal to a specified value. (See DOC M04 for the discussion related to changing the required steam generator water level.) This changes the CTS by moving the procedural details of the specific indicator to be used to verify secondary side water level to the Bases.

The removal of these details for performing surveillance requirements is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of the public health and safety. ITS SR 3.6.4.2 retains the requirement to verify the steam generator water level. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.3.3 states, in part, that required coolant loops shall be verified to be in operation and circulating reactor coolant. ITS SR 3.4.6.1 states, in part, that the required RHR or RCS loop shall be verified to be in operation. This changes the CTS by moving the requirement to verify that the reactor coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a RHR or RCS loop be in operation. This will also require recirculation of reactor coolant, since the ITS Bases specify that verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced or natural circulation flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.4.1.3 ACTION a states, in part, that with less than the above required loops OPERABLE, the unit must be placed in COLD SHUTDOWN within 20 hours. ITS 3.4.6 Required Action A.2 states that when one required loop is inoperable, the unit must be placed in MODE 5 within 24 hours, but only if an RHR loop is OPERABLE. This changes the CTS by providing an exception to the requirements to be in MODE 5 and allowing 24 hours instead of 20 hours to be in MODE 5.

The purpose of CTS 3.4.1.3 ACTION a is to require the unit to be brought to a MODE in which the LCO does not apply. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. The revised actions provide appropriate compensatory measures for an inoperable loop. The CTS requires a cooldown to MODE 5 even if no RHR loops are OPERABLE (i.e., the only OPERABLE loop is an RCS loop.) With only an RCS loop OPERABLE, it is safer to stay in MODE 4 so that the steam generators can be used to remove decay heat. If a cooldown to MODE 5 is required, allowing 24 hours instead of 20 hours is consistent with the times provided in other Specifications, including ITS LCO 3.0.3, to transition from MODE 4 to MODE 5 and is a reasonable time to reach MODE 5 from MODE 4 in an orderly manner and without challenging unit systems. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.4.1.3.1 states, in part, the required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability. ITS SR 3.4.6.3 requires verification of correct breaker alignment and indicated power availability to the pump not in operation

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DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

every 7 days. It is modified by a Note that states "Not required to be performed until 24 hours after a required pump is not in operation." This changes the CTS by not requiring the SR to be performed until 24 hours after a pump is taken out of operation.

The purpose of CTS 4.4.1.3.1 is to ensure that the standby pump is ready to operate. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Note provides time to perform the Surveillance to verify correct breaker alignment and indicated power availability. Without the Note, the Surveillance would not be met immediately after taking a pump out of operation. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.1.3.a
LCO 3.4.1.3.b

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

--NOTES--

LCO 3.4.1.3.b
Footnote **
DOC M02

1. All reactor coolant pumps (RCPs) and RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and "SHUTDOWN MARGIN (SDM)" 5

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.1.3
Footnote *

2. ~~No RCP shall be started with any RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR] unless the secondary side water temperature of each steam generator (SG) is ≤ [50]°F above each of the RCS cold leg temperatures.~~ No reactor coolant pump shall be started unless a steam bubble exists in the pressurizer. 1

Applicability

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND	Immediately

ACTION a
DOC M01

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a DOC L01		A.2 --------------NOTE-------------- Only required if RHR loop is OPERABLE. ------------------------------------- Be in MODE 5.	24 hours
ACTION a ACTION b DOC M01	B. Two required loops inoperable. <u>OR</u> Required loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. <u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.1.3.3	SR 3.4.6.1 Verify required RHR or RCS loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.1.3.2	SR 3.4.6.2	Verify SG secondary side water levels are ≥ ~~[17]~~% 21 for required RCS loops.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 3 3
4.4.1.3.1 DOC L02	SR 3.4.6.3	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.1.3.a
LCO 3.4.1.3.b

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

---NOTES---

LCO 3.4.1.3.b
Footnote **
DOC M02

1. All reactor coolant pumps (RCPs) and RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

    "SHUTDOWN MARGIN (SDM)" 5

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.1.3
Footnote *

2. ~~No RCP shall be started with any RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR] unless the secondary side water temperature of each steam generator (SG) is ≤ [50]°F above each of the RCS cold leg temperatures.~~

No reactor coolant pump shall be started unless a steam bubble exists in the pressurizer. 1

Applicability

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND	Immediately

ACTION a
DOC M01

SEQUOYAH UNIT 2

Amendment XXX

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a DOC L01	A.2 --------------NOTE-------------- Only required if RHR loop is OPERABLE. ------------------------------------- Be in MODE 5.	24 hours
ACTION a ACTION b DOC M01 B. Two required loops inoperable. <u>OR</u> Required loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.4.1.3.3 SR 3.4.6.1 Verify required RHR or RCS loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3 3

SURVEILLANCE REQUIREMENTS (continued)

CTS		SURVEILLANCE	FREQUENCY	
4.4.1.3.2	SR 3.4.6.2	Verify SG secondary side water levels are ≥ ~~[17]~~% for required RCS loops. 21	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 3 3
4.4.1.3.1 DOC L02	SR 3.4.6.3	-------------------------------NOTE------------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6, RCS Loops – MODE 4

1. ISTS LCO 3.4.6 Note 2 has been modified to match CTS 3.4.1.3 Footnote*. CTS 3.4.1.3 Footnote* was approved by the NRC in the Safety Evaluation for License Amendments 147 and 157, Adams Accession Number ML013310424. Therefore SQN will maintain the current licensing requirement for CTS 3.4.1.3 Footnote*.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.4.6.1 (ITS SR 3.4.6.1), ISTS SR 3.4.6.2 (ITS SR 3.4.6.2), and ISTS SR 3.4.6.3 (ITS SR 3.4.6.3) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

4. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. Editorial/grammatical error corrected.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through ~~[~~four~~]~~ RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification. 2

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be OPERABLE ~~available~~ to provide redundancy for decay heat removal. 1

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be

BASES

LCO (continued)

stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

"SHUTDOWN MARGIN",

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, ~~therefore~~ thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation ; and 1 4 4

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that a steam bubble exists in the pressurizer prior to starting any RCP. ~~the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started~~. 3

An OPERABLE RCS loop comprises an OPERABLE RCP and an associated OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2. 1

Similarly for the RHR System, an OPERABLE RHR loop (either A or B) comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. 1

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to ~~meet single failure considerations~~ provide redundant capability of heat removal. 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~;"
LCO 3.4.5, "RCS Loops - MODE 3~~,~~;"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled~~,~~;" 4
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),~~; and 1 4
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 1

ACTIONS

A.1

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and an RHR loop is OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a RHR loop is OPERABLE. With no RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a RHR loop, rather than a cooldown of extended duration.

BASES

ACTIONS (continued)

B.1 and B.2

If two required loops are inoperable or a required loop is not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending operations that would cause the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. 1

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification that the required RCS or RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. 3
~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.~~ 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side ~~narrow range~~ water level is ≥ ~~[17]~~ 21% (narrow range indication). If the SG secondary side ~~narrow range~~ water level is < ~~[17]~~ 21%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ 1 1 2 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 6

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

6

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through ~~[~~four~~]~~ RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification. 2

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be OPERABLE ~~available~~ to provide redundancy for decay heat removal. 1

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be

BASES

LCO (continued)

stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

"SHUTDOWN MARGIN",

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, ~~therefore~~ thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation ; and 1 4 4

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that a steam bubble exists in the pressurizer prior to starting any RCP. ~~the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started~~. 3

An OPERABLE RCS loop comprises an OPERABLE RCP and an associated OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2. 1

Similarly for the RHR System, an OPERABLE RHR loop (either A or B) comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. 1

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to ~~meet single failure considerations~~ provide redundant capability of heat removal. 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~;"
LCO 3.4.5, "RCS Loops - MODE 3~~,~~;"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled~~,~~;" 4
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),~~; and 1 4
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 1

ACTIONS

A.1

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and an RHR loop is OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a RHR loop is OPERABLE. With no RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a RHR loop, rather than a cooldown of extended duration.

BASES

ACTIONS (continued)

B.1 and B.2

If two required loops are inoperable or a required loop is not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending operations that would cause the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. 1

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification that the required RCS or RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. 3
~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.~~

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 6
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side ~~narrow range~~ water level is ≥ ~~[17]~~ 21 % (narrow range indication). If the SG secondary side ~~narrow range~~ water level is < ~~[17]~~ 21 %, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ 1 1 2

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 6

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

6

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6 Bases, RCS LOOPS – MODE 4

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made to be consistent with changes made to the Specifications

4. Editorial/grammatical error corrected.

5. ISTS SR 3.4.6.1 through SR 3.4.6.3 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.6.1 through SR 3.4.6.3 under the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.6, RCS LOOPS – MODE 4

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 7

ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

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

ITS

A01

ITS 3.4.7

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

A03 L01

LCO 3.4.7 3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE~~*~~ and at least one RHR loop shall be in operation.**

Add LCO 3.4.7 Note 4

Add LCO 3.4.7 Note 3 M01

Applicability APPLICABILITY: MODE 5.

with RCS loops filled A02

ACTION:

one M02

ACTION A ACTION B a. With less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible.

Add proposed ACTION C – first paragraph M02

ACTION C b. With no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and immediately initiate corrective action to return ~~the required~~ RHR loop to operation.

OPERABLE status and

one M02

SURVEILLANCE REQUIREMENTS

LA01

SR 3.4.7.1 4.4.1.4 The residual heat removal loop shall be determined to be in operation ~~and circulating reactor coolant at least once per 12 hours~~.

In accordance with the Surveillance Frequency Control Program LA02

LCO 3.4.7 Note 2 # One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

LCO 3.4.7.b Four filled reactor coolant loops with at least 2 steam generators having levels greater than or equal to ~~10 percent (wide-range indication)~~ may be substituted for one RHR loop.

LA03 M03

21% ~~(narrow range indication)~~

~~* The normal or emergency power source may be inoperable.~~ A03

removed from operation A04

per 8 hour period M04

LCO 3.4.7 Note 1 ** The RHR pumps may be ~~de-energized~~ for up to 1 hour provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

Add proposed SR 3.4.7.2 ~~with a Frequency of 12 hours~~ M05

In accordance with the Surveillance Frequency Control Program LA02

Add proposed SR 3.4.7.3 ~~with a Frequency of 7 days~~ M06

ITS

A01

ITS 3.4.7

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

A03

LCO 3.4.7

3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE~~*~~ and at least one RHR loop shall be in operation.**

L01 Add LCO 3.4.7 Note 4

M01 Add LCO 3.4.7 Note 3

Applicability

APPLICABILITY: MODE 5~~.~~ with RCS loops filled A02

ACTION:

ACTION A
ACTION B

a. With one less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible. M02

ACTION C

Add proposed ACTION C – first paragraph M02

b. With no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and immediately initiate corrective action to return ~~the required~~ one RHR loop to OPERABLE status and operation.

M02

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

4.4.1.4 The residual heat removal loop shall be determined to be in operation ~~and circulating reactor coolant at least once per 12 hours~~. LA01

In accordance with the Surveillance Frequency Control Program LA02

LCO 3.4.7 Note 2

LCO 3.4.7.b

\# One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation. Four filled reactor coolant loops with at least 2 steam generators having levels greater than or equal to ~~10 percent (wide-range indication)~~ 21% ~~(narrow range indication)~~ may be substituted for one RHR loop. LA03 M03

~~* The normal or emergency power source may be inoperable.~~ A03

LCO 3.4.7 Note 1

removed from operation A04

per 8 hour period M04

** The RHR pumps may be ~~de-energized~~ removed from operation for up to 1 hour per 8 hour period provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

Add proposed SR 3.4.7.2 ~~with a Frequency of 12 hours~~ M05

In accordance with the Surveillance Frequency Control Program LA02

Add proposed SR 3.4.7.3 ~~with a Frequency of 7 days~~ M06

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.1.4 includes all MODE 5 coolant loop requirements in one Specification. ITS 3.4.7 includes only the MODE 5, Loops Filled requirements. The MODE 5, Loops Not Filled requirements are included in ITS 3.4.8. This changes the CTS by splitting the MODE 5 requirements into two Specifications.

This change is acceptable since all facets of MODE 5 operation are covered in the two ITS Specifications. This change is designated as administrative because it does not result in any technical changes.

A03 CTS 3.4.1.4 footnote * states the residual heat removal (RHR) loops normal or emergency power may be inoperable in MODE 5. ITS 3.4.7 has not retained this specific footnote allowance. This changes the CTS by deleting a specific footnote allowance concerning power supplies. This change is acceptable because the ITS definition of OPERABLE - OPERABILITY requires an OPERABLE component to have only a normal or an emergency power source.

This change to the CTS definition of OPERABLE - OPERABILITY is discussed in the ITS Section 1.0 Discussion of Changes. Given this change to the definition of OPERABLE - OPERABILITY, a specific allowance for the RHR loops is not required. This change is designated as an administrative change since it does not result in a technical change to the CTS.

A04 CTS 3.4.1.4 Footnote ** allows all RHR pumps to be de-energized. ITS LCO 3.4.7 Note 1 allows all RHR pumps to be removed from operation. This changes the CTS by replacing the word "de-energized" to "removed from operation."

The purpose of CTS 3.4.1.4 Footnote ** is to allow the pumps to not meet the requirement of CTS LCO 3.4.1.4 to be in operation. The change better reflects the deviation to the LCO. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.4 states, in part, the number of coolant loops that shall be OPERABLE, and states that at least one RHR loop shall be in operation. ITS 3.4.7 includes Note 3 that states no reactor coolant pump shall be started unless a steam bubble exists in the pressurizer or the secondary side water temperature

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

of each steam generator (SG) is ≤ 25°F above each reactor coolant system (RCS) cold leg temperature. This changes the CTS by adding a restriction for starting a reactor coolant pump.

The purpose of ITS 3.4.7 Note 3 is to prevent a low temperature overpressure event due to thermal transient when a reactor coolant pump is started. This change is acceptable because the restriction ensures a thermal transient will not occur when starting a reactor coolant pump. This change is designated as more restrictive because it adds an explicit requirement for which the CTS does not require.

M02 CTS 3.4.1.4 ACTION a states, in part, that with less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status. CTS 3.4.1.4 ACTION b states, in part, that with no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return the required RHR loop to operation. ITS 3.4.7 ACTION A specifies the Required Actions for one required RHR loop inoperable. The Required Actions are to immediately initiate action to restore a second RHR loop to OPERABLE status and initiate action to restore required SGs secondary side water level to within limit. ITS 3.4.7 ACTION C specifies the Required Actions for no required RHR loops OPERABLE or required RHR loop not in operation. The Required Actions are to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and to initiate action to restore one RHR loop to OPERABLE status and operation. This changes the CTS by adding new immediately Required Actions to be taken when no required RHR loops are OPERABLE.

The purpose of CTS 3.4.1.4 ACTION a is to provide compensatory measures when less than the required RHR/reactor coolant loops are inoperable. CTS 3.4.1.4 ACTION b provides compensatory measures for when no RHR loop is in operation. This change is acceptable because it provides additional actions for the Condition of no required loops OPERABLE. This change is acceptable because it will require immediate actions to preclude boron dilution when no RHR loops are OPERABLE. The additional ITS 3.4.7 ACTION to preclude dilution of the RCS boron concentration provides increased assurance of safe plant operation. This change is designated as more restrictive because it requires additional action to preclude a reduction in the RCS boron concentration when no RHR loops are available for forced circulation of reactor coolant.

M03 CTS 3.4.1.4.a requires at least two residual heat removal (RHR) loops to be OPERABLE and at least one RHR loop to be in operation in MODE 5 with RCS loops filled. Note # allows four filled reactor coolant loops with at least two steam generators having a level greater than or equal to 10 percent (wide range indicator) to substitute for one RHR loop. ITS 3.4.7 requires one RHR loop to be OPERABLE and in operation, and either an additional RHR loop to be OPERABLE or the secondary side of at least two steam generators to be ≥ 21%. ITS SR 3.4.7.2 requires verification that each required steam generator has a secondary side water level ≥ 21% (narrow range indication). (See DOC LA03 for

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

the discussion of moving the procedural details of the specific indicator to be used to verify the secondary side steam generator water level to the Bases.) This changes the CTS by requiring a higher secondary side steam generator water level for determining steam generator OPERABILITY.

The purpose of CTS 3.4.1.4 is to ensure adequate means of decay heat removal and boron mixing are OPERABLE in MODE 5 with the RCS loops filled. When reactor coolant loops are used to meet the requirement, the associated steam generator OPERABILITY is required to be determined by verifying the secondary side water level is high enough to ensure the steam generator tubes are not uncovered, thereby providing the heat sink necessary for decay heat removal. A secondary side steam generator water level of 21% (narrow range indicator) will ensure that at least 76 inches of water is above the top of the steam generators tubes in MODES 1, 2, 3, 4 and 5. This change is acceptable because requiring a secondary side steam generator water level of 21% (narrow range indicator) will ensure the steam generator tubes remain covered. This change is designated as more restrictive because it requires a higher secondary side steam generator water level for steam generator OPERABILITY than was required in CTS.

M04 CTS 3.4.1.4 states that two RHR loops shall be OPERABLE and at least one RHR loop must be in operation. This requirement is modified by Footnote** that states, in part, that all RHR pumps may be de-energized for up to 1 hour. ITS 3.4.7 Note 1 contains the same allowance, but limits the use of the 1 hour exception to once per 8 hours. This changes the CTS by limiting the use of the 1 hour exception to once per 8 hours.

The purpose of the CTS 3.4.1.4 allowance for the RHR pumps to be de-energized for 1 hour is to allow an RHR loop to be removed from operation to support testing. This change is acceptable because it helps ensure that boron stratification and inadequate decay heat removal do not occur by limiting the time in an 8 hour period with no RHR pump in operation. This change is designated as more restrictive because it limits the time the plant can be in MODE 5 with RCS loops filled with no RHR pumps in operation to 1 hour per 8 hour period.

M05 ITS SR 3.4.7.2 requires verification that SG secondary water level is ≥ 10% in required SGs. This Surveillance is not required by the CTS. This changes the CTS by requiring verification that SG secondary water level is ≥ 10% in required SGs when the steam generators are relied upon for decay heat removal.

The purpose of ITS SR 3.4.7.2 is to ensure the required steam generators are available to provide decay heat removal if relied upon to satisfy ITS LCO 3.4.7. This change is acceptable because the verification of proper steam generator level ensures the availability of the steam generators when the steam generators are relied on for decay heat removal, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it adds a Surveillance Requirement to the CTS.

M06 ITS SR 3.4.7.3 requires verification that correct breaker alignment and indicated power are available to each required RHR pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This

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DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required RHR pumps that are not in operation.

The purpose of ITS SR 3.4.7.3 is to ensure a standby RHR pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it adds a Surveillance Requirement to the CTS.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.4 states, in part, that the required residual heat removal loop shall be verified to be in operation and circulating reactor coolant. ITS SR 3.4.7.1 states, in part, that the required RHR loop shall be verified to be in operation. This changes the CTS by moving the words ″circulating reactor coolant″ to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that an RHR loop be in operation. This will require circulation of reactor coolant since the ITS Bases specify that verification that a loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.1.4 requires, in part, verifying the residual heat removal loop to be in operation and circulating coolant at least once per 12 hours. ITS SR 3.4.7.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program. Additionally, ITS SR 3.4.7.2 has been added to verify SG secondary side water level is ≥ 10% in required SGs every 12 hours. ITS SR 3.4.7.3 has been added to verify correct breaker alignment and indicated power are available to each required RHR pump every

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

7 days. (See DOC M05 and M06 for the discussion on adding these SRs.) The "12 hour" and "7 day" Frequencies for these Surveillances have been relocated to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.1.4 Footnote # states, in part, four filled reactor coolant loops with at least 2 steam generators having levels greater than a specified value on the wide range indication may be substituted for one RHR loop. ITS SR 3.4.7.2 has been added (See DOC M05 for the discussion on adding this SR.) and requires, in part, verifying that secondary side water level in each required steam generator greater than a specified value. (See DOC M03 for the discussion related to changing the required steam generator water level.) This changes the CTS by moving the details of the specific indicator used to perform the Surveillance to the Bases.

The removal of the details for performing the Surveillance Requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement to verify that the SG level is ≥ 10%. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.1.4 requires the RHR loops to be OPERABLE and for at least one RHR pump to be operating in MODE 5. ITS 3.4.7 specifies the same requirements; however, ITS LCO 3.4.7 Note 4

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DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS FILLED

allows all RHR loops to be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation. This changes the CTS by adding an allowance for all RHR loops to be removed from operation during planned heatup operations to MODE 4.

The purpose of CTS 3.4.1.4 is to ensure there is sufficient forced circulation to prevent boric acid stratification and to provide forced flow for decay heat removal and transport. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. This change allows an RCS loop to be in operation instead of an RHR loop. The RCS loop simply replaces the function of the RHR loop. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.1.4 — LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

LCO 3.4.1.4 — a. One additional RHR loop shall be OPERABLE or

LCO 3.4.1.4 # — b. The secondary side water level of at least ~~[~~two~~]~~ steam generators (SGs) shall be ≥ ~~[17]~~ 21 %. 1

-----NOTES-----

LCO 3.4.1.4 ** DOC M03 — 1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM "SHUTDOWN MARGIN (SDM)" of LCO 3.1.1; and 5

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.1.4 # — 2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

DOC M01 — 3. No reactor coolant pump shall be started unless a steam bubble exists in the pressurizer or ~~with one or more RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR] unless~~ the secondary side water temperature of each SG is ≤ ~~[50]~~ 25 °F above each of the RCS cold leg temperatures. 2 1

DOC L01 — 4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

Applicability — APPLICABILITY: MODE 5 with RCS Loops Filled.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a	A. One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
ACTION a	B. One or more required SGs with secondary side water level not within limit. AND One RHR loop OPERABLE.	B.1 Initiate action to restore a second RHR loop to OPERABLE status. OR B.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
DOC M02 ACTION b	C. No required RHR loops OPERABLE. OR Required RHR loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.1.4	SR 3.4.7.1	Verify required RHR loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4
DOC M04	SR 3.4.7.2	Verify SG secondary side water level is ≥ ~~[17]~~% in required SGs. 21	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 4
DOC M05	SR 3.4.7.3	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required RHR pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.1.4

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

LCO 3.4.1.4

a. One additional RHR loop shall be OPERABLE or

LCO 3.4.1.4 #

b. The secondary side water level of at least ~~[~~two~~]~~ steam generators (SGs) shall be ≥ ~~[17]~~ 21 %. 1

-------------------------NOTES-------------------------

LCO 3.4.1.4 ** DOC M03

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM "SHUTDOWN MARGIN (SDM)" of LCO 3.1.1; and 5

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.1.4 #

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

DOC M01

3. No reactor coolant pump shall be started unless a steam bubble exists in the pressurizer or ~~with one or more RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR] unless~~ the secondary side water temperature of each SG is ≤ ~~[50]~~ 25 °F above each of the RCS cold leg temperatures. 2 1

DOC L01

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

Applicability

APPLICABILITY: MODE 5 with RCS Loops Filled.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a	A. One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
ACTION a	B. One or more required SGs with secondary side water level not within limit. AND One RHR loop OPERABLE.	B.1 Initiate action to restore a second RHR loop to OPERABLE status. OR B.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
DOC M02 ACTION b	C. No required RHR loops OPERABLE. OR Required RHR loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.1.4	SR 3.4.7.1	Verify required RHR loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4
DOC M04	SR 3.4.7.2	Verify SG secondary side water level is ≥ ~~[17]~~% in required SGs. 21	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 4
DOC M05	SR 3.4.7.3	------NOTE------ Not required to be performed until 24 hours after a required pump is not in operation. ------------ Verify correct breaker alignment and indicated power are available to each required RHR pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS LCO 3.4.7 Note 3 has been modified to match SQN current licensing bases. SQN MODE 5 requirements do not allow starting an RCP without a steam bubble in the pressurizer or the secondary side water temperature of each SG ≤ 25°F above each of the RCS cold leg temperatures. The requirements were approved by the NRC in the Safety Evaluation for License Amendments 147 and 157, Adams Accession Number ML013310424. Therefore SQN will maintain the current licensing requirements.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. ISTS SR 3.4.7.1 (ITS SR 3.4.7.1), ISTS SR 3.4.7.2 (ITS SR 3.4.7.2), and ISTS SR 3.4.7.3 (ITS SR 3.4.7.3) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

5. Editorial/grammatical error corrected.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. 3

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be ~~available~~ OPERABLE to provide redundancy for heat removal. 3

The LCO provides for redundant paths of decay heat removal capability. The first path ~~can be~~ is an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ ~~[17]~~ 21% (narrow range indication) to provide an alternate method for decay heat removal via natural circulation (Ref. 1). 3 4 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ ~~[17]~~ 21% (narrow range indication). One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ ~~[17]~~ 21% (narrow range indication). Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation. 1 4 4 1

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," there~~fore~~ by maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and 1 3 3

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during ~~the only~~ a time when such testing is safe and possible. 3

BASES

LCO (continued)

that a steam bubble exists in the pressurizer or

2
3
2
4

Note 3 requires that the secondary side water temperature of each SG be ≤ ~~[50]~~ °F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) ~~with an RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

25

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least ~~[~~two~~]~~ SGs is required to be ≥ ~~[17]~~ % (narrow range indication).

21

4
4 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"
LCO 3.4.6, "RCS Loops - MODE 4;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),"~~ ; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)."~~

3
2

ACTIONS

A.1, A.2, B.1 and B.2

(narrow range indication)

4 1

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels < ~~[17]~~ % (narrow range indication), or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required

21

BASES

ACTIONS

Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1, or if no required loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal. 3

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification that the required loop is in operation and circulating reactor coolant. 1 Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 6
~~---]~~

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

(narrow range indication)

21 Verifying that at least two SGs are OPERABLE by ensuring their secondary side ~~narrow range~~ water levels are ≥ ~~[17]~~ 21 % (narrow range indication) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ 4 1 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 6

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required RHR pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. If secondary side water level is ≥ ~~[17]~~ 21 % (narrow range indication) in at least two SGs, this Surveillance is not needed. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 4 1 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

6

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. 3

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be ~~available~~ OPERABLE to provide redundancy for heat removal. 3

The LCO provides for redundant paths of decay heat removal capability. The first path is ~~can be~~ an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ ~~[17]~~ 21% (narrow range indication) to provide an alternate method for decay heat removal via natural circulation (Ref. 1). 3 4 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ ~~[17]~~ 21 % (narrow range indication). One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ ~~[17]~~ 21 % (narrow range indication). Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation. 1 4 4 1

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM "SHUTDOWN MARGIN (SDM)," of LCO 3.1.1, there~~fore~~ by maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation ~~and~~ ; 1 3 3

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during ~~the only~~ a time when such testing is safe and possible. 3

BASES

LCO (continued)

that a steam bubble exists in the pressurizer or

Note 3 requires that the secondary side water temperature of each SG be ≤ ~~[50]~~°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) ~~with an RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. 25 2 3 2 4

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least ~~[~~two~~]~~ SGs is required to be ≥ ~~[17]~~% ~~.~~ (narrow range indication) 21 4 4 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"
LCO 3.4.6, "RCS Loops - MODE 4;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),"~~ ; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6).~~" 3 2

ACTIONS

A.1, A.2, B.1 and B.2

(narrow range indication)

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels < ~~[17]~~% ~~,~~ or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required 21 4 1

BASES

ACTIONS

Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1, or if no required loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal. 3

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

and circulating reactor coolant

This SR requires verification that the required loop is in operation. 1
Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ 5

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 6
~~---]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

(narrow range indication)

21 Verifying that at least two SGs are OPERABLE by ensuring their secondary side ~~narrow range~~ water levels are ≥ ~~[17]~~ 21 % (narrow range indication) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. ~~[The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ 4 1

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 6

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required RHR pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. If secondary side water level is ≥ ~~[17]~~ 21 % (narrow range indication) in at least two SGs, this Surveillance is not needed. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 4 1

~~OR~~ 5

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

6

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7 Bases, RCS LOOPS – MODE 5, LOOPS FILLED

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes have been made to be consistent with changes made to the Specifications

3. Editorial/grammatical error corrected.

4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. ISTS SR 3.4.7.1 through SR 3.4.7.3 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.7.1 through SR 3.4.7.3 under the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 8

ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

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

ITS

A01

ITS 3.4.8

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

M01

LCO 3.4.8 3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

Add LCO 3.4.8 Note 1.c

A02

Applicability APPLICABILITY: MODE 5.

with RCS loops not filled

ACTION:

one

M02

ACTION A a. With less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible.

See ITS 3.4.7

Add proposed Condition B, first part

M02

ACTION B b. With no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and immediately initiate corrective action to return the ~~required~~ RHR loop to operation.

one

OPERABLE status and

M02

SURVEILLANCE REQUIREMENTS

LA01

SR 3.4.8.1 4.4.1.4 The residual heat removal loop shall be determined to be in operation ~~and circulating reactor coolant at least once per 12 hours~~.

In accordance with the Surveillance Frequency Control Program

LA02

LCO 3.4.8 Note 2 # One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation. Four filled reactor coolant loops with at least 2 steam generators having levels greater than or equal to 10 percent (wide-range indication) may be substituted for one RHR loop.

See ITS 3.4.7

A03

~~* The normal or emergency power source may be inoperable.~~

removed from operation

A04

LCO 3.4.8 Note 1 Note 1.a Note 1.b ** The RHR pumps may be ~~de-energized for up to 1 hour~~ provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

for ≤ 15 minutes when switching from one loop to another

M03

Add proposed SR 3.4.8.2 with a Frequency of ~~7 days~~

M04

In accordance with the Surveillance Frequency Control Program

LA02

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

DELETED

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ITS A01 ITS 3.4.8

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.4.8 3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.** Add LCO 3.4.8 Note 1.c M01

Applicability APPLICABILITY: MODE 5. with RCS loops not filled A02

ACTION:

one M02

ACTION A a. With less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible. See ITS 3.4.7

Add proposed Condition B, first part M02

ACTION B b. With no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and immediately initiate corrective action to return the ~~required~~ RHR loop to operation.

one OPERABLE status and M02

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 4.4.1.4 The residual heat removal loop shall be determined to be in operation ~~and circulating reactor coolant at least once per 12 hours~~. LA01

In accordance with the Surveillance Frequency Control Program LA02

LCO 3.4.8 Note 2 # One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation. Four filled reactor coolant loops with at least 2 steam generators having levels greater than or equal to 10 percent (wide-range indication) may be substituted for one RHR loop. See ITS 3.4.7

~~* The normal or emergency power source may be inoperable.~~ A03

removed from operation A04

LCO 3.4.8 Note 1 Note 1.a Note 1.b ** The RHR pumps may be ~~de-energized for up to 1 hour~~ provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

for ≤ 15 minutes when switching from one loop to another M03

Add proposed SR 3.4.8.2 with a Frequency of ~~7 days~~ M04

In accordance with the Surveillance Frequency Control Program LA02

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

DELETED

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DISCUSSION OF CHANGES
ITS 3.4.8, MODE 5, RCS LOOPS NOT FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.1.4 includes all MODE 5 coolant loop requirements in one Specification. ITS 3.4.8 includes only the MODE 5 with RCS loops not filled requirements. The MODE 5 with RCS loops filled requirements are included in ITS 3.4.7. This changes the CTS by splitting the MODE 5 requirements into two Specifications.

This change is acceptable since all facets of MODE 5 operation are covered in the two ITS Specifications. This change is designated as administrative because it does not result in any technical changes.

A03 CTS 3.4.1.4 Footnote * states the residual heat removal (RHR) loops normal or emergency power may be inoperable in MODE 5. ITS 3.4.8 has not retained this specific footnote allowance. This changes the CTS by deleting a specific footnote allowance concerning power supplies. This change is acceptable because the ITS definition of OPERABLE - OPERABILITY requires an OPERABLE component to have only a normal or an emergency power source.

This change to the CTS definition of OPERABLE - OPERABILITY is discussed in the ITS Section 1.0 Discussion of Changes. Given this change to the definition of OPERABLE - OPERABILITY, a specific allowance for the RHR loops is not required. This change is designated as an administrative change since it does not result in a technical change to the CTS.

A04 CTS 3.4.1.4 Footnote ** allows all RHR pumps to be de-energized. ITS LCO 3.4.7 Note 1 allows all RHR pumps to be removed from operation. This changes the CTS by changing the words "de-energized" to "removed from operation."

The purpose of CTS 3.4.1.4 Footnote ** is to allow the pumps to not meet the requirement of CTS 3.4.1.4 to be in operation. The change better reflects the deviation to the LCO. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.4 Footnote ** contains an allowance for the RHR pumps to be de-energized for up to one hour provided 1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and 2) core outlet temperature is

DISCUSSION OF CHANGES
ITS 3.4.8, MODE 5, RCS LOOPS NOT FILLED

maintained at least 10°F below saturation temperature. ITS LCO 3.4.8 Note 1 requires all RHR pumps may be removed from operation for 1 hour when switching from one loop to the other provided the core outlet temperature is maintained at least 10°F below saturation temperature, no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.2 and no draining operations to further reduce the RCS water volume are permitted. This changes the CTS 3.4.1.4 Footnote ** by restricting the allowance to only during pump switching operations, and adds a restriction that no draining operations are permitted to further reduce the RCS water volume.

The purpose of the CTS 3.4.1.4 Footnote ** in MODE 5 with RCS loops not filled is to allow the RHR loops to be removed from operation for switching from one loop to the other. This change is acceptable because ITS LCO 3.4.8 Note 1 provides sufficient time to perform loop switching operations and provides adequate controls. Adding the additional condition that no draining operations be performed when the pumps are stopped is reasonable given the low RCS water level and the unavailability of the RHR pumps to add inventory to the RCS, if needed. This change is more restrictive because it imposes additional restrictions to the allowance for RHR loops to be removed from operation.

M02 CTS 3.4.1.4 ACTION a states, in part, with less than the required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status. ITS 3.4.8 ACTION A specifies the Required Action for one required loop inoperable. The Required Action is to immediately initiate action to restore the loop to OPERABLE status. ITS 3.4.8 ACTION B specifies the Required Action when two required loops are inoperable. The Required Action is to immediately suspend operations that would cause introduction of coolant into the RCS with born concentration less than required to meet SDM of LCO 3.1.1, and to initiate action to restore one loop to OPERABLE status and operation. This changes the CTS by adding new immediately Required Actions to be taken when two required loops are inoperable.

The purpose of CTS 3.4.1.4 ACTION a is to provide compensatory measures when one or both required coolant loops are inoperable. This change is acceptable because it provides appropriate actions with two required cooling loops inoperable. Under these conditions, immediate action is necessary to ensure continued safe operation, and action is taken immediately to restore one loop to OPERABLE status to be able to remove decay heat generated by the reactor. This change is designated as more restrictive because it requires immediate action in conditions for which the CTS does not require these actions.

M03 CTS 3.4.1.4 Footnote ** contains an allowance for the RHR pumps to be de-energized for up to one hour. ITS LCO 3.4.8 Note 1 allows all RHR pumps to be removed from operation for ≤ 15 minutes only when switching from one loop to the other, and also requires that no draining operations to further reduce the RCS water volume are permitted (part c). This changes the CTS by reducing the time allowed for the RHR pump to be de-energized from 1 hour to 15 minutes, restricts the allowance to only during pump switching operations, and adds a restriction that no draining operations are permitted to further reduce the RCS water volume.

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DISCUSSION OF CHANGES
ITS 3.4.8, MODE 5, RCS LOOPS NOT FILLED

The purpose of the CTS 3.4.1.4 Footnote ** in MODE 5 with RCS loops not filled is to allow the RHR loops to be removed from operation for switching from one loop to the other. This change is acceptable because ITS LCO 3.4.8 Note 1 provides sufficient time to perform loop switching operations and provides adequate controls. Stopping all operating RHR loops when the RCS is not filled should be limited to short periods of time because of the reduced inventory of water available to absorb decay heat. Stopping all RHR pumps during loop swapping operations may be necessary. Fifteen minutes is sufficient time to perform the loop swapping operation without excessive increases in RCS average temperature due to lack of decay heat removal. Adding the additional condition that no draining operations be performed when the pumps are stopped is reasonable given the low RCS water level and the unavailability of the RHR pumps to add inventory to the RCS, if needed. This change is more restrictive because it reduces the time a RHR loop may be out of service and adds an additional restriction.

M04 ITS SR 3.4.8.2 requires verification that correct breaker alignment and indicated power are available to each required RHR pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required RHR pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required RHR pumps that are not in operation.

The purpose of ITS SR 3.4.8.2 is to ensure a standby RHR pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it adds a Surveillance Requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.4 states, in part, that the required residual heat removal loop shall be verified to be in operation and circulating reactor coolant. ITS SR 3.4.8.1 states, in part, that the required RHR loop shall be verified to be in operation. This changes the CTS by moving the details for verifying RHR pump operation to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that

DISCUSSION OF CHANGES
ITS 3.4.8, MODE 5, RCS LOOPS NOT FILLED

an RHR loop be verified in operation. This will require circulation of reactor coolant since the ITS Bases specify that verification that a loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.1.4 requires, in part, verifying the residual heat removal loop to be in operation and circulating coolant at least once per 12 hours. ITS SR 3.4.8.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program. Additionally, ITS SR 3.4.8.2 has been added to verify correct breaker alignment and indicated power are available to each required RHR pump every 7 days. (See DOC M03 for the discussion on adding the SR.) The "7 day" Frequency for this Surveillance has been moved to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.1.4 — LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

---NOTES---

LCO 3.4.1.4 Note ** — 1. All RHR pumps may be removed from operation for ≤ 15 minutes when switching from one loop to another provided:

LCO 3.4.1.4 Note ** — ~~[~~ a. The core outlet temperature is maintained ~~>~~ ≥ 10°F below saturation temperature~~,~~ ; ~~]~~ 1

LCO 3.4.1.4 Note ** — b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

DOC M01 — c. No draining operations to further reduce the RCS water volume are permitted.

LCO 3.4.1.4 Note # — 2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

Applicability — APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a — A. One required RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M02 ACTION b	B. No required RHR loop OPERABLE. OR Required RHR loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND	Immediately
DOC M02		B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.1.4	SR 3.4.8.1 Verify required RHR loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
DOC M03 SR 3.4.8.2	------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power are available to each required RHR pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.1.4

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

---NOTES---

LCO 3.4.1.4 Note **

1. All RHR pumps may be removed from operation for ≤ 15 minutes when switching from one loop to another provided:

LCO 3.4.1.4 Note **

[a. The core outlet temperature is maintained ~~>~~ ≥ 10°F below saturation temperature~~,~~;] 1

LCO 3.4.1.4 Note **

b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

DOC M01

c. No draining operations to further reduce the RCS water volume are permitted.

LCO 3.4.1.4 Note #

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

Applicability

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a	A. One required RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M02 ACTION b	B. No required RHR loop OPERABLE. OR Required RHR loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
		AND	
DOC M02		B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.4	SR 3.4.8.1	Verify required RHR loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
DOC M03 SR 3.4.8.2	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power are available to each required RHR pump.	~~[7 days~~ ~~OR~~ 3 In accordance with the Surveillance Frequency Control Program ~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.4.8.1 (ITS SR 3.4.8.1), and ISTS SR 3.4.8.2 (ITS SR 3.4.8.2) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be OPERABLE ~~available~~ to provide redundancy for heat removal. 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be removed from operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short ~~[~~and core outlet temperature is maintained ≥ ~~>~~ 10°F below saturation temperature~~]~~. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," is maintained or draining operations when RHR forced flow is stopped. 2 3

BASES

LCO (continued)

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during a ~~the only~~ time when these tests are safe and possible. 5

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~"
LCO 3.4.5, "RCS Loops - MODE 3~~,~~" ;
LCO 3.4.6, "RCS Loops - MODE 4~~,~~" 5
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),~~" and ; 5
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)"~~. 1

ACTIONS

A.1

If one required RHR loop is inoperable, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains 5

BASES

ACTIONS (continued)

acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

and circulating reactor coolant

This SR requires verification that the required loop is in operation. 4
Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 7
~~---]~~

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

7

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be OPERABLE ~~available~~ to provide redundancy for heat removal. 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be removed from operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short ~~[~~and core outlet temperature is maintained ≥ ~~>~~ 10°F below saturation temperature~~]~~. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," is maintained or draining operations when RHR forced flow is stopped. 2

3

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BASES

LCO (continued)

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during a ~~the only~~ time when these tests are safe and possible. 5

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2~~,~~";
LCO 3.4.5, "RCS Loops - MODE 3~~,~~";
LCO 3.4.6, "RCS Loops - MODE 4~~,~~";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled~~,~~"; 5
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" ~~(MODE 6),"~~; and 5 1
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" ~~(MODE 6)"~~.

ACTIONS

A.1

If one required RHR loop is inoperable, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS ~~of coolant~~ with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains 5

BASES

ACTIONS (continued)

acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

and circulating reactor coolant

This SR requires verification that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ 4 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 7

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

7

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8 Bases, RCS LOOPS – MODE 5, LOOPS NOT FILLED

1. Changes have been made to be consistent with the Specification.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The title of the LCO has been provided since this is the first reference to the LCO.

4. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. Editorial/grammatical error corrected.

6. ISTS SR 3.4.8.1 and SR 3.4.8.2 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.8.1 and SR 3.4.8.2 under the Surveillance Frequency Control Program.

7. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 9

ITS 3.4.9, PRESSURIZER

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

ITS A01 ITS 3.4.9

REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

A02 LA01

LCO 3.4.9 3.4.4 The pressurizer shall be OPERABLE with a water level ~~volume of less than or equal to 1656 cubic feet~~ (equivalent to an indicated level of less than or equal to 92% ~~on narrow range instrumentation~~), and at least two groups of pressurizer heaters each having a capacity of at least 150 kw.

Applicability APPLICABILITY: MODES 1, 2 and 3.

ACTION:

ACTION B ACTION C a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

A03 LA02

ACTION A b. With the pressurizer water level not within limit ~~otherwise inoperable~~, be in at least HOT STANDBY ~~with the reactor trip breakers open~~ within 6 hours and in HOT SHUTDOWN within the following 6 hours. Add proposed Required Action A.2 and A.3

SURVEILLANCE REQUIREMENTS

A02 LA03

SR 3.4.9.1 4.4.4.1 The pressurizer water level ~~volume~~ shall be determined to be within its limit ~~at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program.

L02 LA03

SR 3.4.9.2 4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current ~~at least once per 92 days~~ ~~at least once per 18 months~~ In accordance with the Surveillance Frequency Control Program.

L01

~~4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.~~

ITS A01 ITS 3.4.9

REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

A02 level

LCO 3.4.9 3.4.4 The pressurizer shall be OPERABLE with a water ~~volume of less than or equal to 1656 cubic feet~~ (equivalent to an indicated level of less than or equal to 92% ~~on the narrow range instrumentation~~), and at least two groups of presurizer heaters each having a capacity of at least 150 kw. LA01

Applicability APPLICABILITY: MODES 1, 2 and 3

ACTION:

ACTION B
ACTION C
a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY with the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

water level not within limit A03

ACTION A b. With the pressurizer ~~otherwise inoperable~~, be in at least HOT STANDBY ~~with the reactor trip breakers open~~ within 6 hours and in HOT SHUTDOWN within the following 6 hours.

Add proposed Required Action A.2 and A.3 LA02

In accordance with the Surveillance Frequency Control Program LA03

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 4.4.4.1 The pressurizer water ~~volume~~ shall be determined to be within its limit ~~at least once per 12 hours~~.

level A02

SR 3.4.9.2 4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current ~~at least once per 92 days~~.

~~at least once per 18 months~~ L02

In accordance with the Surveillance Frequency Control Program LA03

~~4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.~~ L01

DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.4 requires the pressurizer water volume to be less than or equal to 1656 cubic feet, which is equivalent to an indicated level of less than or equal to 92% of span and CTS 4.4.4.1 requires a verification of the pressurizer water volume. ITS LCO 3.4.9 requires the pressurizer water level to be ≤ 92% and ITS SR 3.4.9.1 requires verification of the pressurizer water level. This changes the CTS by changing "pressurizer water volume" to "pressurizer water level."

The purpose of CTS 3.4.4 and CTS 4.4.4.1 is to ensure the pressurizer water level is at or below the trip setpoint specified in CTS Table 2.2-1. This change is acceptable since the current value corresponds to pressurizer water level. The value of 92% of span corresponds to the Pressurizer Water Level – nominal High trip setpoint in CTS Table 2.2-1. Since the value corresponds to the actual water level in the pressurizer, the change from "volume" to "level" is appropriate. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.4.4 ACTION b applies when the pressurizer is otherwise inoperable (i.e., for reasons other than an inoperable group of pressurizer heaters as described in ACTION a). ITS 3.4.9 ACTION A applies when the pressurizer water level is not within limit. This changes the CTS to specifically state the reason the pressurizer is inoperable.

The purpose of CTS 3.4.4 is to require the pressurizer to be OPERABLE and two conditions of OPERABILITY are supplied. The conditions are pressurizer water level and pressurizer backup heater OPERABILITY. CTS 3.4.4 ACTION b only applies when water level is not within limit. This is the same condition for which ITS 3.4.9 ACTION A applies. This change is acceptable because the condition under which CTS 3.4.4 ACTION b applies has not changed. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

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DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.4 states, in part, that the pressurizer shall be OPERABLE with a water volume of less than or equal to 1656 cubic feet (equivalent to an indicated level of less than or equal to 92% on the narrow range instrumentation). ITS 3.4.9 states, in part, that the pressurizer shall be OPERABLE with a pressurizer water level ≤ 92%. This changes the CTS by moving the details of pressurizer water volume and level indication from the Technical Specification to the Bases

The removal of this detail from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that pressurizer level be ≤ 92%. This change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.4 ACTION b requires the unit to be in at least MODE 3 with the reactor trip breakers open in 6 hours and in MODE 4 within 12 hours if the pressurizer water level limit is not met. Under the same condition, ITS 3.4.9 ACTION A requires the unit to be in MODE 3, to fully insert all rods, and place the Rod Control System in a condition incapable of rod withdrawal within 6 hours. In addition, the unit is required to be in MODE 4 in 12 hours. This changes the CTS by moving the details on how to place the Rod Control System in a condition incapable of rod withdrawal (i.e. by using the reactor trip breakers) from the Technical Specifications to the Bases.

The removal of details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of the public health and safety . The ITS continues to provide actions to exit the mode of applicability. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

LA03 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.4.1 requires verification that pressurizer water volume is within its limit at least once per 12 hours. CTS 4.4.4.2 requires verification of the capacity of the required groups of pressurizer heaters at least once per 92 days. ITS SR 3.4.9.1 and SR 3.4.9.2 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.4.3 requires verification that the emergency power supply for the pressurizer heaters be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters. ISTS SR 3.4.9.3 requires a similar test but does not require the surveillance if pressurizer heater are supplied from a Class 1E power supply. Electrical power to the SQN pressurizer heaters can only be supplied from Class 1E power. This changes the CTS by deleting the Surveillance Requirement to demonstrate OPERABILITY of the pressurizer heater emergency power supply.

The purpose of CTS 4.4.4.3 is to verify OPERABILITY of the pressurizer heater emergency power supply. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the pressurizer heaters used to meet the LCO can perform its required functions. The pressurizer heaters continue to be tested in a manner and at a frequency necessary to give confidence that the pressurizer can perform its assumed safety function. Electrical power to the pressurizer heaters is only provided by Class 1E power sources. Therefore, there is no requirement to verify the transfer from a non-Class 1E power supply to a Class 1E power supply. This change is designated as less restrictive because a Surveillance required in CTS will not be required in ITS.

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DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.4.4.2 requires verification that the capacity of the required pressurizer heaters be verified by measuring circuit current at least once per 92 days. ITS SR 3.4.9.2 requires the same test to be performed at a 18 month Frequency. This changes the CTS by extending the Frequency of the Surveillance from 92 days to 18 months.

The purpose of CTS 4.4.4.2 is to ensure the pressurizer heaters perform as designed. The SQN power supply design for the pressurizer heaters comes from a permanently connected safety related Class 1E power supply. The SQN pressurizer heaters are in constant use, both the proportional and to some extent the backup heaters; therefore SQN does not have any dedicated safety-related pressurizer heaters not used during normal plant operation. ISTS SR 3.4.9.2 Reviewers Note states the frequency for performing the pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied. This change is acceptable because SQN does not have any dedicated safety-related pressurizer heaters not used during normal plant operation. This change is designated as less restrictive because the Surveillance will be performed less frequently under the ITS than CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.4

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ [92]%; and 1 2

b. [Two groups of] pressurizer heaters OPERABLE with the capacity [of each group] ≥ ~~[125]~~ 150 kW ~~[and capable of being powered from an emergency power supply]~~. 1 3

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b A. Pressurizer water level not within limit.	A.1 Be in MODE 3. AND	6 hours
DOC M01	A.2 Fully insert all rods. AND	6 hours
DOC M01	A.3 Place Rod Control System in a condition incapable of rod withdrawal. AND	6 hours
ACTION b	A.4 Be in MODE 4.	12 hours
ACTION a B. One [required] group of pressurizer heaters inoperable.	B.1 Restore [required] group of pressurizer heaters to OPERABLE status.	72 hours 1
ACTION a C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND	6 hours

CTS

Pressurizer
3.4.9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.4.1	SR 3.4.9.1 Verify pressurizer water level is ≤ ~~[~~92~~]~~%.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 5 5
	~~-----REVIEWER'S NOTE----- The frequency for performing Pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~		6
4.4.4.2	SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is ≥ ~~[125]~~ 150 kW.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	5 1 5

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.9.3~~	~~[Verify required pressurizer heaters are capable of being powered from an emergency power supply~~.	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

3

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CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.4

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ [92]%; and 1 2

b. [Two groups of] pressurizer heaters OPERABLE with the capacity [of each group] ≥ ~~[125]~~ 150 kW ~~[and capable of being powered from an emergency power supply]~~. 1 3

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION b	A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours	
		AND			
DOC M01		A.2	Fully insert all rods.	6 hours	
		AND			
DOC M01		A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours	
		AND			
ACTION b		A.4	Be in MODE 4.	12 hours	
ACTION a	B. One [required] group of pressurizer heaters inoperable.	B.1	Restore [required] group of pressurizer heaters to OPERABLE status.	72 hours	1
ACTION a	C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 3.	6 hours	
		AND			

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CTS

Pressurizer
3.4.9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.4.4.1	SR 3.4.9.1 Verify pressurizer water level is ≤ ~~[~~92~~]~~%.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
	~~-----REVIEWER'S NOTE-----~~ ~~The frequency for performing Pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~ ~~----------~~	
4.4.4.2	SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is ≥ ~~[125]~~ 150 kW.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.9.3~~	~~[Verify required pressurizer heaters are capable of being powered from an emergency power supply~~.	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9, PRESSURIZER

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes have been made to correct editorial/grammatical errors.

3. ISTS SR 3.4.9.3 required verifying required pressurizer heaters are capable of being powered from an emergency power supply and is only applicable to pressurizer heaters not powered by Class 1E power supplies. SQN ITS does not include this surveillance. SQN power supply design for the pressurizer heaters can only be powered from a Class 1E power supply and does not provide any other option of power supply. Therefore, this SR is not applicable. ITS 3.4.9 LCO has been changed to reflect the deletion of ISTS SR 3.4.9.3.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. ISTS SR 3.4.9.1 (ITS SR 3.4.9.1) and ISTS SR 3.4.9.2 (ITS SR 3.4.9.2) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

6. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls ~~and emergency power supplies~~. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively. 1

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the U FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. 1

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

~~-----REVIEWER'S NOTE-----~~
~~Plants licensed prior to the issuance of NUREG-0737 may not have a requirement on the number of pressurizer groups.~~
~~----------~~ 2

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ ~~[1240]~~ 1656 cubic feet, which is equivalent to ~~[~~92~~]~~% (narrow range instrumentation), ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. 3 1

The LCO requires ~~[~~two groups of~~]~~ OPERABLE pressurizer heaters, ~~[~~each~~]~~ with a capacity ≥ ~~[125]~~ 150 kW~~, [capable of being powered from either the offsite power source or the emergency power supply]~~. The minimum heater capacity required is sufficient to INSERT 1 ~~maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. [By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [125 kW is derived from the use of seven heaters rated at 17.9 kW each]. The amount needed to maintain pressure is dependent on the heat losses.]~~ 3 1

1 **INSERT 1**

provide assurance that the heaters can be energized during a loss of offsite power condition to provide adequate subcooling margin in the RCS to maintain natural circulation conditions in MODE 3.

BASES

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

. Therefore, they are powered from a Class 1E

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters~~, capable of being powered from an emergency~~ power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable. 1

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level - High Trip Setpoint. 1

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one ~~[~~required~~]~~ group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using ~~normal station powered~~ pressurizer control heaters. 3 1

BASES

ACTIONS (continued)

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level on the narrow range instrumentation. ~~[The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.~~ 1 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 2

SR 3.4.9.2

~~-----REVIEWER'S NOTE-----~~
~~The frequency for performing Pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~
~~-----~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

specified capacity

measuring circuit current and voltage to calculate kW capacity

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their ~~design rating~~. This may be done by ~~testing the power supply output and by performing an electrical check on heater element continuity and resistance~~. ~~[The Frequency of [18] months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ 1 1 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 2

~~[SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. [The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]]~~ 5

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BASES

REFERENCES	1. U FSAR, Section ~~[ ]~~ 15.1.
	2. NUREG-0737, November 1980.

1 3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls ~~and emergency power supplies~~. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively. 1

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the U FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. 1

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

~~-----REVIEWER'S NOTE-----~~
~~Plants licensed prior to the issuance of NUREG-0737 may not have a requirement on the number of pressurizer groups.~~
~~----------~~ 2

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ ~~[1240]~~ 1656 cubic feet, which is equivalent to ~~[~~92~~]~~% (narrow range instrumentation), ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. 3 1

The LCO requires ~~[~~two groups of~~]~~ OPERABLE pressurizer heaters, ~~[~~each~~]~~ with a capacity ≥ ~~[125]~~ 150 kW~~, [capable of being powered from either the offsite power source or the emergency power supply]~~. The minimum heater capacity required is sufficient to INSERT 1 ~~maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. [By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [125 kW is derived from the use of seven heaters rated at 17.9 kW each]. The amount needed to maintain pressure is dependent on the heat losses.]~~ 3 1

1 **INSERT 1**

provide assurance that the heaters can be energized during a loss of offsite power condition to provide adequate subcooling margin in the RCS to maintain natural circulation conditions in MODE 3.

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BASES

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

. Therefore, they are powered from a Class 1E

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters~~, capable of being powered from an emergency~~ power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable. 1

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level - High Trip Setpoint. 1

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one ~~[~~required~~]~~ group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using ~~normal station powered~~ pressurizer control heaters. 3 1

BASES

ACTIONS (continued)

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level on the narrow range instrumentation. ~~[The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.~~ 1 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 2

SR 3.4.9.2

~~-----REVIEWER'S NOTE-----~~
~~The frequency for performing Pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~
~~-----~~ 2

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BASES

SURVEILLANCE REQUIREMENTS (continued)

specified capacity

measuring circuit current and voltage to calculate kW capacity

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their ~~design rating~~. This may be done by ~~testing the power supply output and by performing an electrical check on heater element continuity and resistance~~. ~~[The Frequency of [18] months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ 1 1 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 2

~~[SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. [The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]]~~ 5

BASES

REFERENCES

1. U FSAR, Section ~~[ ]~~ 15.1. 1 3

2. NUREG-0737, November 1980.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9 Bases, PRESSURIZER

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.4.9.1 and ISTS SR 3.4.9.2 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.9.1 and ITS SR 3.4.9.2 under the Surveillance Frequency Control Program.

5. ISTS SR 3.4.9.3 is not included in the SQN ITS since the surveillance only applies to those pressurizer heaters not powered by Class 1E power supplies. SQN pressurizer heaters are powered from a Class 1E power. Therefore, this SR is not applicable.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.9, PRESSURIZER

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ITS 3.4.10, PRESSURIZER SAFETY VALVES

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

ITS

A01

ITS 3.4.10

REACTOR COOLANT SYSTEM

3/4.4.3 SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

≥ 2410 psig and ≤ 2559 psig

A02

LCO 3.4.10 3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of ~~2485 PSIG ± 3%~~.*

Applicability APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Add proposed Applicability Note

L01

ACTION A a. With one pressurizer safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes.

ACTION B b. With two or more pressurizer safety valves inoperable or with ACTION (a) above not completed within 15 minutes, be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1 4.4.3.1 No additional Surveillance Requirements other than those required by ~~Specification 4.0.5~~. Following testing, lift settings shall be within ± 1%.

the Inservice Testing Program

A03

LA01

~~* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure~~.

ITS

A01

ITS 3.4.10

REACTOR COOLANT SYSTEM

3/4.4.3 SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

≥ 2410 psig and ≤ 2559 psig

A02

LCO 3.4.10

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of ~~2485 PSIG ± 3%~~.*

Applicability

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Add proposed Applicability Note

L01

ACTION A

a. With one pressurizer safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes.

ACTION B

b. With two or more pressurizer safety valves inoperable or with ACTION (a) above not completed within 15 minutes, be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

4.4.3.1 No additional Surveillance Requirements other than those required by ~~Specification 4.0.5~~. Following testing, lift settings shall be within ± 1%.

the Inservice Testing Program

A03

LA01

~~* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.~~

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DISCUSSION OF CHANGES
ITS 3.4.10, PRESSURIZER SAFETY VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.3.1 requires all pressurizer code safety valves to be OPERABLE with a lift setting of 2485 psig ± 3%. ITS LCO 3.4.10 requires three pressurizer safety valves to be OPERABLE with lift settings ≥ 2410 psig and ≤ 2559 psig. This changes the CTS by including the actual lift settings, in lieu of a plus and minus tolerance band.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.4.10 now provides the actual lift settings in lieu of a tolerance band. The ITS lift settings of ≥ 2410 psig and ≤ 2559 psig are the CTS tolerance band of 2485 psig ± 3%. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 4.4.3.1 requires no additional Surveillance Requirements on the pressurizer safety valves other than those required by Specification 4.0.5. ITS SR 3.4.10.1 requires verification that each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program with a Frequency of in accordance with the Inservice Testing Program. This changes the CTS by stating pressurizer safety valve testing is performed in accordance with the Inservice Testing Program, and that the Frequency is in accordance with the Inservice Testing Program.

The purpose of CTS 4.4.3.1 is to verify each pressurizer safety valve is tested in accordance with Specification 4.0.5, which provides the requirements for the Inservice Testing Program. This change is acceptable, because the Frequency regarding the pressurizer safety valves testing remains the same. The inservice testing requirements of CTS 4.0.5 have been moved to the Inservice Testing Program contained in Section 5.5 of the ITS. This change is designated as administrative, because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.4.10, PRESSURIZER SAFETY VALVES

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.3.1 is modified by a footnote that states that the pressurizer lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. This information is not provided in ITS 3.4.10. This changes the CTS by moving this information from the Technical Specifications to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.10 still retains a requirement for the valves to be OPERABLE. Under the definition of OPERABILITY, the safety valves must be capable of lifting at the assumed conditions, which include the ambient operating conditions of the safety valves themselves. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being moved from the Technical Specifications to the ITS Bases.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.4.3.1, in part, provides requirements for the pressurizer code safety valves. ITS LCO 3.4.10 Applicability is modified by a Note that allows the lift settings to not be within the LCO limits during MODE 3 for the purpose of in-situ setting of the pressurizer safety valves under ambient (hot) conditions. The exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. This changes the CTS by allowing entry into MODE 3 without verifying that the pressurizer code safety valve lift settings are within the LCO limits.

The purpose of the Applicability Note is to allow entry into MODE 3 to perform testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. This change is acceptable because the requirements continue to ensure that the components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The cold lift settings give assurance that the valves are OPERABLE near their design condition during the short period of time allowed to verify the settings at the hot condition. While SQN does not set pressurizer safety valves while installed at this time, this Applicability Note provides the flexibility to utilize this method in the future. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.3.1 | LCO 3.4.10 | ~~[~~Three~~]~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~[2460]~~ 2410 psig and ≤ ~~[2510]~~ 2559 psig. 1

Applicability | APPLICABILITY: | MODES 1, 2, and 3~~;~~.
~~MODE 4 with all RCS cold leg temperatures > [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR].~~ 2

DOC L01

---NOTE---
The lift settings are not required to be within the LCO limits during MODE~~S~~ 3 ~~and 4~~ for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ~~[~~54~~]~~ hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. 2 1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
ACTION b	B. Required Action and associated Completion Time of Condition A not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 ~~with any RCS cold leg temperatures ≤ [275°F] [LTOP arming temperature specified in the PTLR]~~.	6 hours ~~[24]~~ 12 hours 2 1

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.3.1	SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

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CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.3.1

LCO 3.4.10 ~~[~~Three~~]~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~[2460]~~ 2410 psig and ≤ ~~[2510]~~ 2559 psig. 1

Applicability

APPLICABILITY: MODES 1, 2, and 3~~;~~.
~~MODE 4 with all RCS cold leg temperatures > [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR].~~ 2

DOC L01

---NOTE---
The lift settings are not required to be within the LCO limits during MODE~~S~~ 3 ~~and 4~~ for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ~~[~~54~~]~~ hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. 2 1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes	
ACTION b	B. Required Action and associated Completion Time of Condition A not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 ~~with any RCS cold leg temperatures ≤ [275°F] [LTOP arming temperature specified in the PTLR]~~.	6 hours ~~[24]~~ 12 hours	2 1

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.3.1	SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. As provided in SQN Units 1 and 2 License Amendments 299 and 288, respectively, Adams Accession Number ML 050480669, dated March 9, 2005, CTS 3.4.2, "Safety Valves Shutdown," was eliminated because overpressure protection of the reactor coolant system (RCS) does not rely upon the pressurizer safety valves during plant operation in MODES 4 and 5. For SQN, the PTLR provides an LTOP arming temperature of 350°F. This temperature coincides with the MODE 3 to 4 transition temperature. In MODE 4 the LTOP System is enabled and designed to protect the RCS from overpressure conditions during low temperature plant conditions. As such, ISTS 3.4.10 Applicability, the Note modifying the applicability and Required Action B.2 have been changed to reflect the SQN licensing basis that requires the pressurizer safety valves to be OPERABLE in MODES 1, 2, and 3.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), ~~[~~2735~~]~~ psig, which is 110% of the design pressure. 1

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, ~~[380,000]~~ 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level. 1

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, ~~with one or more RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR], and~~ MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." 2 4

The upper and lower pressure limits are based on the ± ~~1~~ 3% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3 (nominal operating temperature and pressure). This requires either that the valves be set hot or that a correlation between hot and cold settings be established. 3

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE SAFETY ANALYSES

The most limiting ~~All~~ accident and safety analyses in the U FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of ~~[~~three~~]~~ safety valves. Accidents that could result in overpressurization if not properly terminated include: 3

a. Uncontrolled rod withdrawal from full power,

b. Loss of reactor coolant flow, ~~(reactor coolant pump locked rotor)~~ ~~3~~

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c. Loss of external electrical load,

d. Loss of normal feedwater, ~~and~~ ~~4~~

e. Loss of all AC power to station auxiliaries, and ~~, and~~ ~~.~~ ~~4~~

f. Locked rotor. ~~f. Locked rotor.~~ ~~3~~

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, and e (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The ~~[~~three~~]~~ pressurizer safety valves are set to open at the RCS design pressure (2485 psig ~~2500 psia~~), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 3 ~~1~~% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation. 1 3

APPLICABILITY

In MODES 1, 2, and 3~~, and portions of MODE 4 above the LTOP arming temperature,~~ OPERABILITY of ~~[~~three~~]~~ valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is ~~and portions of MODE 4 are~~ conservatively included, although the listed accidents may not require the safety valves for protection. 2 1 2

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 ~~when any RCS cold leg temperatures are ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~ or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned. 2

The Note allows entry into MODE~~S~~ 3 ~~and 4~~ with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The ~~[~~54~~]~~ hour exception is based on 18 hour outage time for each of the ~~[~~three~~]~~ valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe. 2 1

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 ~~with any RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~ within 12 ~~[24]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ~~With any RCS cold leg temperatures at or below [275°F] [Low Temperature Overpressure (LTOP) arming temperature specified in the PTLR]~~ In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by ~~[~~three~~]~~ pressurizer safety valves. 2 1 2 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

of 2485 psig 3 1

The pressurizer safety valve setpoint is ± ~~[~~3~~]~~% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. U FSAR, Chapter ~~[~~15~~]~~. 3 1
3. WCAP-7769, Rev. 1, June 1972.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), ~~[~~2735~~]~~ psig, which is 110% of the design pressure. 1

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, ~~[380,000]~~ 420,000 lb/hr, is based on postulated overpressure transient 1 conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, ~~with one or more RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR], and~~ MODE 5, and MODE 6 with the reactor vessel 2 4 head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± ~~1~~ 3% tolerance 3 requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3 (nominal operating temperature and pressure). This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

The most limiting U

APPLICABLE SAFETY ANALYSES

~~All~~ accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of ~~[~~three~~]~~ safety valves. Accidents that could result in overpressurization if not properly terminated include: 3

a. Uncontrolled rod withdrawal from full power,

b. Loss of reactor coolant flow, ~~(reactor coolant pump locked rotor)~~ 3

RPG-012

c. Loss of external electrical load,

d. Loss of normal feedwater, ~~and~~ , and 4

e. Loss of all AC power to station auxiliaries~~, and~~ . 4

f. Locked rotor. ~~f. Locked rotor.~~ 3

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, and e (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The ~~[~~three~~]~~ pressurizer safety valves are set to open at the RCS design pressure (2485 psig ~~2500 psia~~), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 3 ~~1~~% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation. 1 3

APPLICABILITY

In MODES 1, 2, and 3~~, and portions of MODE 4 above the LTOP arming temperature,~~ OPERABILITY of ~~[~~three~~]~~ valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is ~~and portions of MODE 4 are~~ conservatively included, although the listed accidents may not require the safety valves for protection. 2 1 2

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 ~~when any RCS cold leg temperatures are ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~ or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned. 2

The Note allows entry into MODE~~S~~ 3 ~~and 4~~ with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The ~~[~~54~~]~~ hour exception is based on 18 hour outage time for each of the ~~[~~three~~]~~ valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe. 2 1

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 ~~with any RCS cold leg temperatures ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]~~ within 12 ~~[24]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ~~With any RCS cold leg temperatures at or below [275°F] [Low Temperature Overpressure (LTOP) arming temperature specified in the~~ In MODE 4 ~~PTLR]~~, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by ~~[~~three~~]~~ pressurizer safety valves. 2 1 2 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

of 2485 psig

The pressurizer safety valve setpoint is ± ~~[~~3~~]~~% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. 3 1

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. U FSAR, Chapter ~~[~~15~~]~~. 3 1
3. WCAP-7769, Rev. 1, June 1972.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10 Bases, PRESSURIZER SAFETY VALVES

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes have been made to reflect changes made to the Specification.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. Editorial/grammatical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

ITS 3.4.11, PRESSURIZER POWER OPERATED RELEIF VALVES (PORVs)

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

ITS A01 ITS 3.4.11

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.4.11 3.4.3.2 Two power relief valves (PORVs) and their associated block valves shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, and 3.

Add proposed ACTIONS Note. A02

ACTION:

being manually cycled A03

ACTION A a. With one or more PORV(s) inoperable, but capable of ~~RCS pressure control~~, within 1 hour ~~either restore the PORV(s) to OPERABLE status or~~ close the associated block valve(s); otherwise, be in at least HOT STANDBY within the

ACTION D next 6 hours and in HOT SHUTDOWN within the following 6 hours.

A04

with power maintained A05

being manually cycled A03

b. With one PORV inoperable and incapable of ~~RCS pressure control~~, within 1 hour ~~either restore the PORV to OPERABLE status or~~ close the associated

ACTION B block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within

ACTION D the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

A04

Add proposed Required Action Note C.1. A06

c. With both PORVs inoperable and incapable of ~~RCS pressure control~~, within 1 hour ~~either restore each of the PORVs to OPERABLE status or~~ close their

ACTION E associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

being manually cycled A03

A04

Add proposed Required Action Note F.1. A06

ACTION C

ACTION F d. With one or more block valve(s) inoperable, within 1 hour: (1) restore ~~the~~ block valve~~(s)~~ to OPERABLE status~~, or close the block valve(s) and remove power from the block valve(s), or close the PORV(s) and remove power from its associated solenoid valve(s); and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s)~~.

ACTION C

one M01

Place associated PORV in manual control L03

Add proposed Required Action C.2 L01

SURVEILLANCE REQUIREMENTS

A07

SR 3.4.11.2 4.4.3.2.1 ~~In addition to the requirements of Specification 4.0.5~~, each PORV shall be demonstrated OPERABLE ~~at least once per 18 months~~ by:

In accordance with the Surveillance Frequency Control Program LA01

a. DELETED

SR 3.4.11.2
SR 3.4.11.2 Note b. Operating the valve through one complete cycle of full travel during Mode 3, 4, or 5 with a steam bubble in the pressurizer.

Add proposed SR 3.4.11.1 Notes 1 and 2 L02

SR 3.4.11.1 4.4.3.2.2 Each block valve shall be demonstrated OPERABLE ~~at least once per 92 days~~ by operating the valve through one complete cycle of full travel.

In accordance with the Surveillance Frequency Control Program LA01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.3.2.3 (Deleted.)

~~4.4.3.2.4 In addition to the requirements of Specification 4.0.5 the repair welds and adjoining areas of the pressurizer relief line shall be examined, using improved UT detection and evaluation procedures which have been demonstrated to be effective in detecting IGSCC, prior to entering MODE 4 whenever the plant has been in COLD SHUTDOWN for 72 hours or more if the examination has not been performed in the previous 6 months.~~

~~In the event these 6-month period examinations find the piping free of unacceptable indications for 3 successive inspections, the inspection interval shall be extended to 36 month intervals (± 12 months to coincide with a scheduled refueling outage).~~

~~In the event these 36-month period examinations find the piping free of unacceptable indications for 3 successive inspections, the inspection interval shall be extended to 80-month periods.~~

LA02

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ITS A01 ITS 3.4.11

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.4.11 3.4.3.2 All power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, and 3.

Add proposed ACTIONS Note. A02

ACTION:

being manually cycled A03

ACTION A a. With one or more PORV(s) inoperable, but capable of ~~RCS pressure control~~, within 1 hour ~~either restore the PORV(s) to OPERABLE status or~~ close the associated block valve(s); otherwise, be A04

ACTION D in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

with power maintained A05

being manually cycled A03

b. With one PORV inoperable and incapable of ~~RCS pressure control~~, within 1 hour ~~either restore the PORV to OPERABLE status or~~ close the associated block valve and remove power from the A04

ACTION B block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT A06

ACTION D STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Add proposed Required Action Note C.1.

being manually cycled A03

c. With both PORVs inoperable and incapable of ~~RCS pressure control~~, within 1 hour ~~either restore each of the PORVs to OPERABLE status or~~ close their associated block valves and remove A04

ACTION E power from the block valves and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

ACTION C Add proposed Required Action Note F.1. A06

ACTION F d. With one or more block valve(s) inoperable, within 1 hour: (1) restore ~~the~~ one block valve~~(s)~~ to OPERABLE status~~, or close the block valve(s) and remove power from the block valve(s), or close the PORV(s) and remove power from its associated solenoid valve(s); and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s)~~. M01

ACTION C Place associated PORV in manual control. L03

Add proposed Required Action C.2 L01

SURVEILLANCE REQUIREMENTS

SR 3.4.11.2 4.4.3.2.1 ~~In addition to the requirements of Specification 4.0.5,~~ each PORV shall be demonstrated OPERABLE ~~at least once per 18 months by~~: A07

In accordance with the Surveillance Frequency Control Program LA01

a. DELETED

SR 3.4.11.2
SR 3.4.11.2 Note b. Operating the valve through one complete cycle of full travel during Modes 3, 4, or 5 with a steam bubble in the pressurizer.

Add proposed SR 3.4.11.1 Note 1 and 2 L02

SR 3.4.11.1 4.4.3.2.2 Each block valve shall be demonstrated OPERABLE ~~at least once per 92 days~~ by operating the valve through one complete cycle of full travel.

In accordance with the Surveillance Frequency Control Program LA01

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.3.2 ACTIONS a, b, c, and d describe the compensatory actions to take when PORV(s) and/or block valve(s) are inoperable. ITS 3.4.11 ACTIONS A, B, C, D, E, F, and G also state the appropriate compensatory actions under the same conditions, however, an ITS 3.4.11 ACTIONS Note has been added. The ITS 3.4.11 ACTION Note allows separate Condition entry for each Pressurizer PORV and PORV block valve. This changes the CTS by providing a specific allowance to enter the Action for each inoperable Pressurizer PORV and PORV block valve.

The purpose of the Note is to provide explicit instructions for proper application of the Action for Technical Specification compliance. In conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing Action for inoperable Pressurizer PORVs and PORV block valves. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.4.3.2 ACTION a applies to one or more PORVs inoperable but capable of RCS pressure control. CTS 3.4.3.2 ACTIONS b and c apply to one or both PORVs inoperable and incapable of RCS pressure control. ITS 3.4.11 ACTION A applies to one or more PORVs inoperable and capable of being manually cycled. ITS 3.4.11 ACTION B applies to one PORV inoperable and not capable of being manually cycled. This changes the CTS by replacing the word "pressure control" with "being manually cycled".

This change is acceptable because the requirements have not changed. An inoperable PORV that is capable of RCS pressure control can still be manually cycled. PORVs inoperable for other reasons and incapable of pressure control cannot be manually cycled. Therefore, the conditions under which the Required Actions are applied have not changed. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.4.3.2 ACTIONS a, b, and c provide compensatory measures for inoperable PORV(s) and provide an alternative option to restore inoperable PORV(s) to OPERABLE status. ITS 3.4.11 ACTIONS A, B, and E provide similar compensatory measures, but do not include the explicit option to restore the valves to OPERABLE status. This changes the CTS by deleting the alternative Action to restore the valves to OPERABLE status.

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DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

This change is acceptable because the requirements have not changed. ITS LCO 3.0.2 states that upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. Therefore, based on ITS LCO 3.0.2, restoration is always an option. This change is designated as administrative as it does not result in a technical change to the CTS.

A05 CTS 3.4.3.2 ACTION a states, in part, with one or more PORV(s) inoperable but capable of RCS pressure control, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s). ITS 3.4.11 ACTION A specifies the Required Action for one or more PORVs inoperable and capable of being manually cycled to close and maintain power to associated block valve. This changes the CTS by explicitly stating that power to the closed block valve is to be maintained.

The purpose of CTS 3.4.3.2 ACTION a is to isolate the inoperable PORV with an OPERABLE block valve that retains the capability to be manually operated for pressure control. This change is acceptable because the requirements have not changed. CTS 3.4.3.2 ACTION a does not specify to remove power from the associated block valve being closed. Therefore, the conditions under which the Required Actions are applied in ITS have not changed. This change is designated as administrative because it does not result in a technical change to the CTS.

A06 CTS 3.4.3.2 ACTION d specifies the compensatory actions for one or more inoperable block valves. ITS 3.4.11 ACTION C specifies the Required Actions for one inoperable block valve and ITS 3.4.11 ACTION F specifies the Required Actions for two inoperable block valves. The ITS 3.4.11 ACTIONS C and F Required Actions are preceded by a Note that states the specified Required Actions (C.1, C.2 and F.1) do not apply when the block valve is inoperable solely as a result of complying with Required Action B.2 or E.2. ITS 3.4.11 Required Actions B.2 and E.2 require the removal of power from the applicable block valve when a PORV is inoperable and incapable of being manually cycled. This changes the CTS by adding the clarification Note that the Required Action to place the PORV in manual control is not applicable when the block valve is inoperable solely due to complying with the ACTIONS for an inoperable PORV.

The purpose of the CTS 3.4.3.2 Actions is to ensure the appropriate compensatory measures are taken with inoperable PORVs or inoperable block valves. The Note clarifies that the applicable Required Action is not necessary when entry into the Condition is made as a result of application of the Required Actions for inoperable PORVs that are not capable of being manually cycled. This clarification is acceptable since these actions (place associated PORV in manual control or restore one block valve to OPERABLE status) are not appropriate for the block valve inoperability. This change is designated as administrative since the change does not result in a technical change to the CTS.

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DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

A07 CTS 4.4.3.2.1 states that the PORVs must be tested in accordance with Specification 4.0.5, the Inservice Testing Program requirements for ASME Code Class 1, 2, and 3 components. ITS 3.4.11 does not contain this explicit Surveillance Requirement. This changes the CTS by deleting the explicit requirement to perform the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components.

The purpose of CTS 4.4.3.2.1 is to ensure the appropriate inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components are performed for the required PORVs. The inservice testing requirements of CTS 4.0.5 are retained in ITS 5.5.6, "Inservice Testing Program." See the Discussion of Changes for ITS 5.5 for any changes to the requirements of CTS 4.0.5. The explicit cross reference is not necessary. If the system is determined to be inoperable when tested in accordance with the Inservice Testing Program, the plant procedures will require the PORVs to be declared inoperable and the appropriate ITS 3.4.11 ACTIONS will be entered when applicable. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.3.2 ACTION d requires with one or more inoperable block valves to, within 1 hour, either restore the block valve(s) to an OPERABLE status, close the block valve and remove power from the block valve, or close the associated PORV and remove power from it associated solenoid valve. ITS 3.4.11 ACTION C provides actions for one block valve inoperable and requires the associated PORV be placed in manual control in 1 hour and the block valve to be restored to an OPERABLE status in 72 hours. ITS 3.4.11 ACTION F provides actions for two block valves inoperable and requires restoration of one block valve to OPERABLE status within 1 hour. This changes the CTS by deleting the option to close and remove power from the block valve as a compensatory action for an inoperable block valve. (See DOC L01 for a discussion of extending the time to restore one inoperable block valve from 1 hour to 72 hours.)

The purpose of CTS 3.4.3.2 ACTION d is to provide the appropriate compensatory actions for when one or more block valves are inoperable. A block valve is inoperable when it is incapable of either being energized opened with the capability to close, or energized closed with the capability to open. In most cases, the option to close the block valve will not be available if the block valve is inoperable such that it cannot be closed. Therefore, the action to place the associated PORV in manual control is acceptable for one block valve being inoperable. Placing the PORV in manual control prevents the PORV from opening automatically for an overpressure event and avoids the potential for a stuck open PORV at a time when its associated block valve is inoperable. This change is designated as more restrictive as it deletes an optional compensatory action.

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DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.3.2.1.b requires, in part, verification that each PORV can be operated through one complete cycle of full travel once per 18 months. ITS SR 3.4.11.2 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." CTS 4.4.3.2.2 requires, in part, verification that each block valve can be operated through one complete cycle of full travel once per 92 days. ITS SR 3.4.11.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies of these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 4.4.3.2.4 (Unit 1) states, in part, the repair welds and adjoining areas of the pressurizer relief line shall be examined. ITS 3.4.11 does not contain Inservice Inspection Program requirements. This changes the CTS by removing these requirements from the Technical Specifications to the Inservice Inspection Program.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled in the Inservice Inspection Program, which is controlled under 10 CFR 50.55a. This change is

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

designated as a less restrictive removal of detail change because the ISI testing requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.3.2 ACTION d requires, in part, with one or more block valve(s) inoperable, within 1 hour (1) restore the block valve (s) to operable status. ITS 3.4.11 ACTION C requires, in part, with one block valve inoperable to restore block valve to OPERABLE status within 72 hours. This changes the CTS by extending the time to restore the inoperable block valve from one hour to 72 hours.

The purpose of CTS 3.4.3.2 ACTION d is to provide compensatory measures for an inoperable PORV block valve. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of the remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The time allowed to restore the block valve is based on the Completion Time for restoring an inoperable PORV in Condition B. This change is designated as less restrictive because additional time is allowed to restore one PORV block valve to OPERABLE status than was allowed in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.4.3.2.2 requires each block valve to be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. ITS SR 3.4.11.1 requires a similar surveillance and adds two Notes. Note 1 states the Surveillance is not required to be performed with a block valve closed in accordance with the Required Actions of the LCO. Note 2 states the Surveillance is only required to be performed in MODES 1 and 2. This changes the CTS by not requiring a cycle of the block valve when the block valve is closed due to excessive PORV leakage and by not requiring performance of the Surveillance prior to entry into MODE 3.

The purpose of CTS 4.4.3.2.2 is to verify the block valve can be cycled, if needed. ITS SR 3.4.11.1 Note 1 allows the Surveillance to not be performed when the block valve is closed in accordance with the Required Actions of the LCO. The purpose of the Note is to prevent opening of the block valve in a condition that increases the risk of an unisolabled leak from the RCS since the PORV is already inoperable. ITS SR 3.4.11.2 Note 2 modifies the Surveillance to allow entry into and operation in MODE 3 prior to the performance of the Surveillance. The Note allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODES 1 and 2. Therefore, this change is acceptable because it has been determined that opening a block valve used to isolate an inoperable PORV increases the risk of an unisolable leak from the RCS. Also, allowance for the entry into MODE 3 for the performance of the Surveillance is acceptable based on the test is being performed under operating and pressure conditions. These changes are

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

designated as less restrictive because the Surveillance will be performed less frequently under the ITS than under the CTS.

L03 *(Category 4 – Relaxation of Required Action)* CTS 3.4.3.2 ACTION d requires, in 1 hour, when one or more block valve(s) are inoperable to either restore the block valve(s), close and remove power from the block valve(s), or close the PORV(s) and remove power from its associated solenoid valve(s), and declare the associated PORV(s) inoperable and enter appropriate ACTIONS for the isolated PORV(s). ITS 3.4.11 ACTION C applies for the Condition of one inoperable block valve and requires within 1 hour, the associated PORV to be placed in manual control. This changes the CTS by requiring the associated PORV to be placed in manual control instead of closing and removing power from its associated solenoid valve and does not require the associated PORV(s) to be declared inoperable. (See DOC L01 for a discussion of extending the time to restore an inoperable block valve. See DOC M01 for a discussion of removing the option of closing and removing power from the inoperable block valve.)

The purpose of CTS 3.4.3.2 ACTION d is to provide appropriate compensatory actions for an inoperable block valve. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems and features. This includes the capacity and capability of the remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. When a block valve is inoperable the associated PORV is closed. The primary purpose of a block valve is to isolate a stuck open PORV. If the PORV is placed in manual control it cannot be opened inadvertently; therefore, the primary purpose of the block valve is being met. The PORV still has the capability to be manually opened. Placing the PORV in manual control does not prevent the PORV from performing its primary function of depressurizing the RCS in response to certain transients if normal pressurizer spray is not available. Therefore, it is not necessary to remove power from its associated PORV to prevent inadvertently opening the PORV. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.3.2

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

DOC A02

---NOTE---
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a — A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
ACTION b — B. One ~~[or two]~~ PORV~~[s]~~ inoperable and not capable of being manually cycled.	B.1 Close associated block valve~~[s]~~.	1 hour
	AND	
	B.2 Remove power from associated block valve~~[s]~~.	1 hour
	AND	
	B.3 Restore PORV~~[s]~~ to OPERABLE status.	72 hours

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CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION d DOC A06	C. One ~~[or two]~~ block valve~~(s)~~ inoperable.	--------------------NOTE-------------------- Required Actions C.1 and C.2 do not apply when block valve is inoperable solely as a result of complying with Required Action~~s~~ B.2 or E.2. --		2 3 3
		C.1 Place associated PORV in manual control.	1 hour	
		AND		
DOC L01		C.2 Restore block valve to OPERABLE status.	72 hours	
ACTION a ACTION b ACTION d	D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours	
		AND		
		D.2 Be in MODE 4.	12 hours	
ACTION c	E. Two ~~[or three]~~ PORVs inoperable and not capable of being manually cycled.	E.1 Close associated block valves.	1 hour	2
		AND		
		E.2 Remove power from associated block valves.	1 hour	
		AND		
		E.3 Be in MODE 3.	6 hours	
		AND		
		E.4 Be in MODE 4.	12 hours	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
ACTION d DOC A06 — F. Two ~~[or three]~~ block valves inoperable.	F.1	--------------NOTE-------------- Required Action F.1 does not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 ------------------------------------ Restore one block valve to OPERABLE status ~~[if three block valves are inoperable]~~.	1 ~~[2~~ hour~~s]~~ 2
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Be in MODE 4.	12 hours

2

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
4.4.3.2.2 DOC L02	SR 3.4.11.1	-----------------------------NOTES---------------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4
4.4.3.2.1 4.4.3.2.1.b	SR 3.4.11.2	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 4
	~~SR 3.4.11.3~~	~~[Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	5

SEQUOYAH UNIT 1 Amendment XXX

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
~~SR 3.4.11.4 [Verify PORVs and block valves are capable of being powered from emergency power sources.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

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CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.3.2

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

DOC A02

-----NOTE-----
Separate Condition entry is allowed for each PORV and each block valve.

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour	
ACTION b	B. One ~~[or two]~~ PORV~~[s]~~ inoperable and not capable of being manually cycled.	B.1 Close associated block valve~~[s]~~.	1 hour	2
		AND		
		B.2 Remove power from associated block valve~~[s]~~.	1 hour	2
		AND		
		B.3 Restore PORV~~[s]~~ to OPERABLE status.	72 hours	2

CTS

Pressurizer PORVs
3.4.11

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION d DOC A06	C. One ~~[or two]~~ block valve~~(s)~~ inoperable.	--------------------NOTE------------------- Required Actions C.1 and C.2 do not apply when block valve is inoperable solely as a result of complying with Required Action~~s~~ B.2 or E.2. ---		2 3 3
		C.1 Place associated PORV in manual control.	1 hour	
		AND		
DOC L01		C.2 Restore block valve to OPERABLE status.	72 hours	
ACTION a ACTION b ACTION d	D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours	
		AND		
		D.2 Be in MODE 4.	12 hours	
ACTION c	E. Two ~~[or three]~~ PORVs inoperable and not capable of being manually cycled.	E.1 Close associated block valves.	1 hour	2
		AND		
		E.2 Remove power from associated block valves.	1 hour	
		AND		
		E.3 Be in MODE 3.	6 hours	
		AND		
		E.4 Be in MODE 4.	12 hours	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
ACTION d DOC A06 F. Two ~~[or three]~~ block valves inoperable.	F.1	--------------NOTE-------------- Required Action F.1 does not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 ------------------------------------ Restore one block valve to OPERABLE status ~~[if three block valves are inoperable]~~.	1 ~~[2~~ hour~~s]~~ 2
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.3.2.2 DOC L02	SR 3.4.11.1	-----------------------------NOTES---------------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 4
4.4.3.2.1 4.4.3.2.1.b	SR 3.4.11.2	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 4
	~~SR 3.4.11.3~~	~~[Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ 5

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.11.4~~	~~[Verify PORVs and block valves are capable of being powered from emergency power sources.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Typographical error corrected and editorial change made for enhanced clarity.

4. ISTS SR 3.4.11.1 (ITS SR 3.4.11.1), and ISTS SR 3.4.11.2 (ITS SR 3.4.11.2) provide two options for controlling the Frequencies of the Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

5. The bracketed requirement is deleted because it is not applicable to SQN. The PORVs control system design does not contain solenoid air control valves, check valves, and air accumulators.

6. The bracketed requirement is deleted because it is not applicable to SQN. The PORVs and block valves are not configured such that they can be powered from a non-safety related power supply.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

solenoid

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are ~~air~~ operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room. 1

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the block valves during power operation. 2

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

BASES

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A Note ~~1~~ has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). 3

~~REVIEWER'S NOTE~~
~~The bracketed options in Conditions B, C, E, and F are to accommodate plants with three PORVs and associated block valves.~~ 4

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, ~~either the PORVs must be restored or the flow path isolated within 1 hour.~~ ~~T~~ t he associated block valve is required to be closed within 1 hour, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. 3 ~~This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.~~ 5

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

If one ~~[or two]~~ PORV~~[s]~~ is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. 6

C.1 and C.2

If one ~~[or two]~~ block valve~~(s)~~ is ~~are~~ inoperable, then it is necessary to either restore the block valve(s) to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve~~(s)~~ is to isolate a stuck open PORV. Therefore, if the block valve(s) cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve~~(s)~~ is ~~are~~ inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve~~(s)~~ to OPERABLE status. The time allowed to restore the block valve~~(s)~~ is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve~~(s)~~ Is not ~~are not~~ full open. If the block valve~~(s)~~ is ~~are~~ restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. 6 3 3 3 3

BASES

ACTIONS (continued)

The Required Actions C.1 and C.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3, and E.4

If two ~~more than one~~ PORVs are ~~is~~ inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12. 3

BASES

ACTIONS (continued)

F.1

If two ~~[or three]~~ block valve(s) are inoperable, it is necessary to restore at 6
least one block valve within ~~2~~ 1 hour~~s~~. The Completion Time is reasonable, 3
based on the small potential for challenges to the system during this time
and provide s the operator time to correct the situation. 7

Required Action F.1 is modified by a Note stating that the Required Action does not apply if the sole reason for the block valve being declared inoperable is a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

G.1 and G.2

If the Required Action of Condition F is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. ~~[The basis for the Frequency of 92 days is the ASME Code (Ref. 3).~~ 8

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[~~In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.~~]~~ 6 6

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. ~~[The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~ 8

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[~~In accordance with Reference 4, administrative controls require this test be performed in MODE 3, ~~or~~ 4 or 5 with a steam bubble in the pressurizer to adequately simulate operating temperature and pressure effects on PORV operation.~~]~~ 6 1 6

~~[SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. [The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]]~~ 6

~~[SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~

~~The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
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REFERENCES

1. Regulatory Guide 1.32, February 1977.

U

2. FSAR, Section ~~[~~15.2~~]~~.

1 6

3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~[~~ 4. ~~Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.]~~

Letter from R.W. Henan (NRC Staff) to J.A. Scalice (TVA), "Issuance of Technical Specification Amendments for Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA2164 and MA2169) (TS 98-01," dated November 19, 1998 (ADAMS Accession No. ML013320468).

6

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

solenoid

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are ~~air~~ operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room. 1

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation. 2

block

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

BASES

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

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BASES

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A Note ~~1~~ has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). 3

~~-----REVIEWER'S NOTE-----~~
~~The bracketed options in Conditions B, C, E, and F are to accommodate plants with three PORVs and associated block valves.~~
~~-----~~ 4

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, ~~either the PORVs must be restored or the flow path isolated within 1 hour.~~ ~~T~~he associated block valve is required to be closed within 1 hour, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. ~~This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.~~ t 3 5

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

If one ~~[or two]~~ PORV~~[s]~~ is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. 6

C.1 and C.2

If one ~~[or two]~~ block valve~~(s)~~ is ~~are~~ inoperable, then it is necessary to either restore the block valve(s) to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve~~(s)~~ is to isolate a stuck open PORV. Therefore, if the block valve(s) cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve~~(s)~~ is ~~are~~ inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve~~(s)~~ to OPERABLE status. The time allowed to restore the block valve~~(s)~~ is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve~~(s)~~ Is not ~~are not~~ full open. If the block valve~~(s)~~ is ~~are~~ restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. 6 3 3 3 3

BASES

ACTIONS (continued)

The Required Actions C.1 and C.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3, and E.4

If two ~~more than one~~ PORVs are ~~is~~ inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12. 3

BASES

ACTIONS (continued)

F.1

If two ~~[or three]~~ block valve(s) are inoperable, it is necessary to restore at least one block valve within ~~2~~ 1 hour~~s~~. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. 6 3 7

Required Action F.1 is modified by a Note stating that the Required Action does not apply if the sole reason for the block valve being declared inoperable is a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

G.1 and G.2

If the Required Action of Condition F is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. ~~[The basis for the Frequency of 92 days is the ASME Code (Ref. 3).~~ 8

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[~~In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.~~]~~ 6 6

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. ~~[The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~ 8

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[~~In accordance with Reference 4, administrative controls require this test be performed in MODE 3, ~~or~~ 4 or 5 with a steam bubble in the pressurizer to adequately simulate operating temperature and pressure, effects on PORV operation.~~]~~ 6 1 6

~~[SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. [The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]]~~

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~~[SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~

~~The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
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REFERENCES

1. Regulatory Guide 1.32, February 1977.

U

2. FSAR, Section ~~[~~15.2~~]~~. 1 6

3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~[~~ 4. ~~Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.]~~

Letter from R.W. Henan (NRC Staff) to J.A. Scalice (TVA), "Issuance of Technical Specification Amendments for Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA2164 and MA2169) (TS 98-01," dated November 19, 1998 (ADAMS Accession No. ML013320468).

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11 BASES, PRESSURIZER POWER OPERATED VALVES (PORVS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS (and ITS) SR for PORV cycling (SR 3.4.11.2) only requires the test every 18 months. Furthermore, NUREG-1316 emphasizes the important of not stroking the PORVs during power operation. Therefore, the word “valves” has been replaced with “block valves” to clarify that only block valves can be tested online.

3. Changes are made to be consistent with the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. This statement has been deleted since the statement is not valid. The required Action does not preclude the unit from starting up without performing the maintenance on the valve(s).

6. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

7. Editorial changes made for enhanced clarity.

8. ISTS SR 3.4.11.1 and ISTS 3.4.11.2 (ITS SR 3.4.11.1 and ITS 3.4.11.2) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.4.11.1 and ITS 3.4.11.2 is "In accordance with the Surveillance Frequency Control Program."

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.11, PRESSURIZER POWER OPERATED RELIEF VALVES (PORVs)

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 12

ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

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

REACTOR COOLANT SYSTEM

3/4.4.12 LOW TEMPERATURE OVER PRESSURE PROTECTION (LTOP) SYSTEM

LIMITING CONDITION FOR OPERATION

LA01

LCO 3.4.12 3.4.12* An LTOP System shall be OPERABLE with a maximum of one ~~centrifugal~~ charging pump capable of injecting into the Reactor Coolant System (RCS) and the accumulators isolated and one of the following pressure relief capabilities:

and no safety injection pump A02

LCO 3.4.12.a a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

LCO 3.4.12.b b. The RCS depressurized and an RCS vent ≥ 3 square inches.

Applicability APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ the LTOP arming temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTION:

ACTION A
ACTION B
ACTION A ACTION B

a. Should any safety injection pump or more than one charging pump be found capable of injecting into the RCS, immediately initiate action to verify a maximum of one ~~centrifugal~~ charging pump is capable of injecting into the RCS.

LA01

ACTION C b. With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR, isolate the affected accumulator within 1 hour, or either;

ACTION D

1. Increase RCS cold leg temperature to > the LTOP arming temperature specified in the PTLR within 12 hours, or

2 Depressurize the affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR within 12 hours.

ACTION E c. With one required PORV inoperable in MODE 4, restore the required PORV to OPERABLE status within 7 days.

ACTION F d. With one required PORV inoperable in MODE 5 or 6, restore the required PORV to OPERABLE status within 24 hours.

LCO 3.4.12 NOTE 1 * 1) Two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations.

LCO 3.4.12 NOTE 2 2) Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

LCO 3.4.12 NOTE 3 3) For the purpose of making the required safety injection pumps and charging pump inoperable, the following time is permitted: up to 4 hours after entering MODE 4 from MODE 3, or prior to decreasing temperature on any RCS loop to below 325°F, whichever occurs first.

REACTOR COOLANT SYSTEM

ACTION (Continued)

ACTION G e. With two required PORVs inoperable, or the Actions (a), (b), (c) , or (d) not met, or the LTOP System inoperable for any reason other than (a), (b), (c), or (d), depressurize the RCS and establish RCS vent of ≥ 3.0 square inches within 12 hours.

ACTIONS NOTE f. LCO 3.0.4.b is not applicable when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.12.1 Each PORV shall be demonstrated OPERABLE by:

SR 3.4.12.6 a. Performance of a CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST*, but excluding valve operation, ~~at least once per 31 days;~~ In accordance with the Surveillance Frequency Control Program (M01, LA02)

SR 3.4.12.7 b. Performance of a CHANNEL CALIBRATION on each required PORV actuation channel ~~at least once per 18 months; and~~ In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.5 c. Verifying the PORV block valve is open for each required PORV ~~at least once per 72 hours~~. In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.1 4.4.12.2 Verify no safety injection pumps are capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F, ~~and every 12 hours thereafter~~. In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.2 4.4.12.3 Verify a maximum of one charging pump is capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F, ~~and every 12 hours thereafter~~. In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.3 4.4.12.4 Verify each accumulator is isolated ~~at least once per 12 hours~~. In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.4 4.4.12.5 Verify# required RCS vent ≥ 3.0 square inches open ~~at least~~:

a. ~~Once every 12 hours for unlocked open vent valve(s) and,~~ In accordance with the Surveillance Frequency Control Program (LA02)

b. ~~Once every 31 days for other vent path(s).~~ In accordance with the Surveillance Frequency Control Program (LA02)

SR 3.4.12.6 NOTE * Not required to be performed until 12 hours after decreasing RCS cold leg temperatures to ≤ the LTOP arming temperature in the PTLR.

SR 3.4.12.4 NOTE # Only required to be met when complying with LCO 3.4.12.b.

ITS A01 ITS 3.4.12

REACTOR COOLANT SYSTEM

3/4.4.12 LOW TEMPERATURE OVER PRESSURE PROTECTION (LTOP) SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.4.12 3.4.12* An LTOP System shall be OPERABLE with a maximum of one ~~centrifugal~~ charging pump and no safety injection pump capable of injecting into the Reactor Coolant System (RCS) and the accumulators isolated and one of the following pressure relief capabilities: LA01 A02

LCO 3.4.12.a a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

LCO 3.4.12.b b. The RCS depressurized and an RCS vent ≥ 3 square inches.

Applicability APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ the LTOP arming temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTION:

ACTION A
ACTION B
ACTION A
ACTION B

a. Should any safety injection pump or more than one charging pump be found capable of injecting into the RCS, immediately initiate action to verify a maximum of one ~~centrifugal~~ charging pump is capable of injecting into the RCS. LA01

ACTION C b. With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR, isolate the affected accumulator within 1 hour, or either;

ACTION D

1. Increase RCS cold leg temperature to >the LTOP arming temperature specified in the PTLR within 12 hours, or

2 Depressurize the affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR within 12 hours.

ACTION E c. With one required PORV inoperable in MODE 4, restore the required PORV to OPERABLE status within 7 days.

ACTION F d. With one required PORV inoperable in MODE 5 or 6, restore the required PORV to OPERABLE status within 24 hours.

LCO 3.4.12 NOTE 1 * 1) Two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations.

LCO 3.4.12 NOTE 2 2) Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

LCO 3.4.12 NOTE 3 3) For the purpose of making the required safety injection pumps and charging pump inoperable, the following time is permitted: up to 4 hours after entering MODE 4 from MODE 3, or prior to decreasing temperature on any RCS loop to below 325°F, whichever occurs first.

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ITS

A01

ITS 3.4.12

REACTOR COOLANT SYSTEM

ACTION (Continued)

ACTION G

e. With two required PORVs inoperable, or the Actions (a), (b), (c) , or (d) not met, or the LTOP System inoperable for any reason other than (a), (b), (c), or (d), depressurize the RCS and establish RCS vent of ≥ 3.0 square inches within 12 hours.

ACTIONS NOTE

f. LCO 3.0.4.b is not applicable when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.12.1 Each PORV shall be demonstrated OPERABLE by:

SR 3.4.12.6

a. Performance of a CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST*, but excluding valve operation, ~~at least once per 31 days~~ In accordance with the Surveillance Frequency Control Program; M01 LA02

SR 3.4.12.7

b. Performance of a CHANNEL CALIBRATION on each required PORV actuation channel ~~at least once per 18 months; and~~ In accordance with the Surveillance Frequency Control Program LA02

SR 3.4.12.5

c. Verifying the PORV block valve is open for each required PORV ~~at least once per 72 hours~~ In accordance with the Surveillance Frequency Control Program. LA02

SR 3.4.12.1

4.4.12.2 Verify no safety injection pumps are capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 and prior to the temperature of one or more RCS cold legs decreasing below 325°F, ~~and every 12 hours thereafter~~ In accordance with the Surveillance Frequency Control Program. LA02

SR 3.4.12.2

4.4.12.3 Verify a maximum of one charging pump is capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 and prior to the temperature of one or more RCS cold legs decreasing below 325°F, ~~and every 12 hours thereafter~~ In accordance with the Surveillance Frequency Control Program. LA02

SR 3.4.12.3

4.4.12.4 Verify each accumulator is isolated ~~at least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program LA02

SR 3.4.12.4

4.4.12.5 Verify# required RCS vent ≥ 3.0 square inches open ~~at least:~~

a. ~~Once every 12 hours for unlocked open vent valve(s) and~~ In accordance with the Surveillance Frequency Control Program, LA02

b. ~~Once every 31 days for other vent path(s)~~ In accordance with the Surveillance Frequency Control Program LA02

SR 3.4.12.6 NOTE

* Not required to be performed until 12 hours after decreasing RCS cold leg temperatures to ≤ the LTOP arming temperature in the PTLR.

SR 3.4.12.4 NOTE

\# Only required to be met when complying with LCO 3.4.12.b.

DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.12 requires LTOP to be OPERABLE with a maximum of one centrifugal charging pump capable of injecting into the Reactor Coolant System (RCS), with the accumulators isolated, and with one of the listed pressure relief capabilities. CTS 3.4.12 ACTION a requires immediate action to verify a maximum of one charging pump capable of injecting into the RCS should any safety injection pump be found capable of injecting into the RCS. CTS 4.4.12.2 requires verification that no safety injection pump is capable of injecting into the RCS. ITS LCO 3.4.12 requires LTOP to be OPERABLE with a maximum of one charging pump and no safety injection pump capable of injecting into the RCS, with the accumulators isolated, and with one of the listed pressure relief capabilities. This changes CTS by explicitly including in the LCO statement that LTOP OPERABILITY is dependent on no safety injection pumps being capable of injecting into the RCS.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.4.12 includes the requirement that no safety injection pump is capable of injecting into the RCS. ITS SR 3.4.2.1 requires the verification that no safety injection pumps are capable of injection into the RCS. This change is acceptable because the addition of the requirement in ITS LCO 3.4.12 that no safety injection pump be capable of injecting into the RCS is implied by the CTS Surveillance Requirement. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.4.12.1.a requires, in part, each PORV be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST. ITS SR 3.4.12.6 requires performance of a CHANNEL OPERATIONAL TEST (COT) to demonstrate PORV operability. This changes the CTS by changing the CHANNEL FUNCTIONAL TEST requirements to a COT.

CTS defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated signal into the sensor as close to the sensor as practicable to verify OPERABILITY. ITS defines a COT as the injection of an actual or simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the

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DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

setpoints are within the necessary range and accuracy. This changes the CTS by requiring adjustments of the setpoints so that each PORV Channel is within the necessary range and accuracy. This change is designated as more restrictive because it imposes additional requirements on testing.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.12 states, in part, that the LTOP System shall be OPERABLE with a maximum of one centrifugal charging pump capable of injecting into the Reactor Coolant System (RCS). CTS 3.4.12 ACTION a states, in part, to immediately initiate action to verify a maximum of one centrifugal charging pump is capable of injecting into the RCS. ITS 3.4.12 states, in part, that the LTOP System shall be OPERABLE with a maximum of one charging pump capable of injecting into the RCS. This changes the CTS by moving the details related to the type of charging pump to the ITS Bases.

The removal of these details which are related to the design of the charging pump from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection to public health and safety. ITS 3.4.12 still retains the requirement that a maximum of one charging pump is OPERABLE and capable of injecting into the RCS. Also, this change is acceptable because these types of design details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because design details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.12.1.a states each PORV shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST but excluding valve operation at least once per 31 days. ITS SR 3.4.12.6 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.1.b states each PORV shall be demonstrated OPERABLE by performance of a CHANNEL CALIBRATION on each required PORV actuation channel at least once per 18 months. ITS SR 3.4.12.7 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.1.c states each PORV shall be demonstrated OPERABLE by verifying the PORV block valve is open for each required PORV at least once per 72 hours. ITS SR 3.4.12.5 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.2 states to verify no safety injection pumps are capable of

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DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one RCS cold legs decreasing below 325°F and every 12 hours thereafter. ITS SR 3.4.12.1 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.3 states to verify a maximum of one charging pump is capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one RCS cold legs decreasing below 325°F and every 12 hours thereafter. ITS SR 3.4.12.2 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.4 states to verify each accumulator is isolated at least once per 12 hours. ITS SR 3.4.12.3 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." CTS 4.4.12.5.a and b state to verify required RCS vent ≥ 3.0 square inches open at least once every 12 hours for unlocked open vent valves and once every 31 days for other vent path(s). ITS SR 3.4.12.4 requires a similar Surveillance and specifies a periodic Frequency of "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for the SRs to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

and no safety injection pump

3

LCO 3.4.12

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of ~~[one] [high pressure injection (HPI)] pump [and~~ one charging pump~~]~~ capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities: 1

LCO 3.4.12.a

a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR~~,~~ ; or

~~[b. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig,]~~ 2

~~[c. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig,] or~~

LCO 3.4.12.b

~~d~~ b. The RCS depressurized and an RCS vent of ≥ ~~[2.07]~~ 3.0 square inches. 2 1

---NOTES---

LCO 3.4.12.* 1)

1. ~~[~~Two charging pumps~~]~~ may be made capable of injecting for ≤ 1 hour for pump swap operations. 1

LCO 3.4.12.* 2)

2. Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

LCO 3.4.12.* 3)

3. Two safety injection pumps and two charging pumps may be capable of injecting for ≤ 4 hours after entering MODE 4 from MODE 3 or prior to lowering temperature on any RCS loop below 325°F, whichever occurs first. 3

Applicability

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, 1
MODE 5,
MODE 6 when the reactor vessel head is on.

CTS

LTOP System
3.4.12

ACTIONS

ACTION f

---NOTE---
LCO 3.0.4.b is not applicable when entering MODE 4.

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. ~~Two~~ One or more ~~[HPI]~~ safety injection pumps capable of injecting into the RCS.	A.1 Initiate action to verify ~~a maximum of [one] [HPI]~~ no safety injection pumps are ~~is~~ capable of injecting into the RCS.	Immediately	1 4 1
ACTION a	B. [Two ~~or more~~ charging pumps capable of injecting into the RCS.	B.1 Initiate action to verify a maximum of [one] charging pump is capable of injecting into the RCS.	Immediately]	1
ACTION b	C. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	C.1 Isolate affected accumulator.	1 hour	
ACTION b.1 ACTION b.2	D. Required Action and associated Completion Time of Condition [C] not met.	D.1 Increase RCS cold leg temperature to > ~~[275°F]~~ [LTOP arming temperature specified in the PTLR].	12 hours	1
		OR		
		D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours	
ACTION c	E. One required ~~RCS relief valve~~ PORV inoperable in MODE 4.	E.1 Restore required ~~RCS relief valve~~ PORV to OPERABLE status.	7 days	4

ACTINS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION d	F. One required PORV ~~RCS relief valve~~ inoperable in MODE 5 or 6.	F.1 Restore required PORV ~~RCS relief valve~~ to OPERABLE status.	24 hours	4
ACTION e	G. Two required PORVs ~~RCS relief valves~~ inoperable. OR Required Action and associated Completion Time of Condition A, ~~[~~B,~~]~~ D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, ~~[~~B,~~]~~ C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ ~~[2.07]~~ 3.0 square inches.	12 hours	4 1 1 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.12.2	SR 3.4.12.1 Verify no safety injection ~~a maximum of [one] [HPI]~~ pumps are ~~is~~ capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ INSERT 1 In accordance with the Surveillance Frequency Control Program ~~]~~	4 1 5 3 5

3 **INSERT 1**

4.4.12.2 Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F.

AND

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SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
4.4.12.3	SR 3.4.12.2	[Verify a maximum of one charging pump is capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ INSERT 2 In accordance with the Surveillance Frequency Control Program]]	1 5 3 1 5
4.4.12.4	SR 3.4.12.3	Verify each accumulator is isolated.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	5 5
	~~SR 3.4.12.4~~	~~[Verify RHR suction valve is open for each required RHR suction relief valve.~~	~~[12 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

3 **INSERT 2**

4.4.12.3 Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F.

AND

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.12.5	SR 3.4.12.~~5~~ 4	INSERT 3 Verify required RCS vent ≥ ~~[2.07]~~ 3.0 square inches open.	~~[12 hours for unlocked open vent valve(s)~~ ~~AND~~ ~~31 days for other vent path(s)~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	6 2 1 5 5
4.4.12.1.c	SR 3.4.12.~~6~~ 5	Verify PORV block valve is open for each required PORV.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 5
	~~SR 3.4.12.7~~	~~[Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

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6 **INSERT 3**

-----------------------------NOTE--------------------------------

4.4.12.5 # Only required to be met when complying with LCO 3.4.12.b.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.12.1.a 4.4.12.1.a*	SR 3.4.12.~~8~~ 6	------------------------------NOTE----------------------------- Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275°F]~~ ~~[~~LTOP arming temperature specified in the PTLR~~]~~. --- Perform a COT on each required PORV, excluding actuation.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 5 5
4.4.12.1.b	SR 3.4.12.~~9~~ 7	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 5

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

and no safety injection pump (3)

LCO 3.4.12 — LCO 3.4.12 — An LTOP System shall be OPERABLE with a maximum of ~~[one] [high pressure injection (HPI)] pump [and~~ one charging pump~~]~~ capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities: (1)

LCO 3.4.12.a — a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR~~,~~ ; or

~~[b. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig,]~~

~~[c. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig,] or~~ (2)

LCO 3.4.12.b — ~~d~~ b. The RCS depressurized and an RCS vent of ≥ ~~[2.07]~~ 3.0 square inches. (2) (1)

--NOTES--

LCO 3.4.12.* 1) — 1. ~~[~~Two charging pumps~~]~~ may be made capable of injecting for ≤ 1 hour for pump swap operations. (1)

LCO 3.4.12.* 2) — 2. Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

LCO 3.4.12.* 3) — 3. Two safety injection pumps and two charging pumps may be capable of injecting for ≤ 4 hours after entering MODE 4 from MODE 3 or prior to lowering temperature on any RCS loop below 325°F, whichever occurs first. (3)

Applicability — APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, (1)
MODE 5,
MODE 6 when the reactor vessel head is on.

CTS

ACTIONS

ACTION f

-----NOTE-----
LCO 3.0.4.b is not applicable when entering MODE 4.

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. One ~~Two~~ or more safety injection ~~[HPI]~~ pumps capable of injecting into the RCS.	A.1 Initiate action to verify no safety injection ~~a maximum of [one] [HPI]~~ pumps are ~~is~~ capable of injecting into the RCS.	Immediately	1 4 1
ACTION a	B. [Two ~~or more~~ charging pumps capable of injecting into the RCS.	B.1 Initiate action to verify a maximum of [one] charging pump is capable of injecting into the RCS.	Immediately]	1
ACTION b	C. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	C.1 Isolate affected accumulator.	1 hour	
ACTION b.1 ACTION b.2	D. Required Action and associated Completion Time of Condition [C] not met.	D.1 Increase RCS cold leg temperature to > ~~[275°F]~~ [LTOP arming temperature specified in the PTLR].	12 hours	1
		OR		
		D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours	
ACTION c	E. One required PORV ~~RCS relief valve~~ inoperable in MODE 4.	E.1 Restore required PORV ~~RCS relief valve~~ to OPERABLE status.	7 days	4

CTS

LTOP System
3.4.12

ACTINS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION d	F. One required PORV ~~RCS relief valve~~ inoperable in MODE 5 or 6.	F.1 Restore required PORV ~~RCS relief valve~~ to OPERABLE status.	24 hours	4
ACTION e	G. Two required PORVs ~~RCS relief valves~~ inoperable. OR Required Action and associated Completion Time of Condition A, [B,] D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, [B,] C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ 3.0 ~~[2.07]~~ square inches.	12 hours	4 1 1 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.12.2	SR 3.4.12.1	Verify no safety injection ~~a maximum of [one] [HPI]~~ pumps are ~~is~~ capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ INSERT 1 In accordance with the Surveillance Frequency Control Program]	4 1 5 3 5

3 **INSERT 1**

4.4.12.2 Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F.

AND

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.12.3	SR 3.4.12.2	[Verify a maximum of one charging pump is capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ INSERT 2 In accordance with the Surveillance Frequency Control Program]]	1 5 3 1 5
4.4.12.4	SR 3.4.12.3	Verify each accumulator is isolated.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	5 5
	~~SR 3.4.12.4~~	~~[Verify RHR suction valve is open for each required RHR suction relief valve.~~	~~[12 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

3 **INSERT 2**

4.4.12.3 Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F.

AND

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CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.12.5	SR 3.4.12.~~5~~ 4	INSERT 3 Verify required RCS vent ≥ ~~[2.07]~~ 3.0 square inches open.	~~[12 hours for unlocked open vent valve(s)~~ ~~AND~~ ~~31 days for other vent path(s)~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	6 2 1 5 5
4.4.12.1.c	SR 3.4.12.~~6~~ 5	Verify PORV block valve is open for each required PORV.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 5
	~~SR 3.4.12.7~~	~~[Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

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6 **INSERT 3**

4.4.12.5 #

-----------------------------NOTE--------------------------------
Only required to be met when complying with
LCO 3.4.12.b.

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.12.1.a 4.4.12.1.a*	SR 3.4.12.~~8~~ 6	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275°F]~~ ~~[~~LTOP arming temperature specified in the PTLR~~]~~. --- Perform a COT on each required PORV, excluding actuation.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 5 5
4.4.12.1.b	SR 3.4.12.~~9~~ 7	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 5 5

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. References to residual heat removal (RHR) suction relief valves are being deleted from the ISTS 3.4.12. RHR suction relief valves are not required to be disabled in SQN Units 1 and 2 LTOP analysis. Due to this change, subsequent ACTIONS and SRs have been renumbered. This change is also consistent with SQN CTS.

3. A Note has been added to LCO 3.4.12 that states "Two safety injection and two charging pumps may be capable of injecting into the RCS for ≤ 4 hours after entering MODE 4 from MODE 3 prior to lowering temperature on any RCS loop below 325°F." Surveillance Frequencies have been added to ITS SR 3.4.12.1 and ITS SR 3.4.12.2 that require "Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F". This change is consistent with SQN CTS.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. ISTS SR 3.4.12.1 (ITS SR 3.4.12.1), ISTS SR 3.4.12.2 (ITS SR 3.4.12.2) and ISTS SR 3.4.12.3 (ITS SR 3.4.12.3) provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for this SR under the Surveillance Frequency Control Program.

6. A Note has been added to ITS SR 3.4.12.4 "Only required to be met when complying with LCO 3.4.12.b." This change is consistent with SQN CTS.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but ~~[one] [high pressure injection (HPI)] pump [and~~ one centrifugal charging pump~~]~~ incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs ~~RCS relief valves~~ or a depressurized RCS and an RCS vent of sufficient size. One PORV ~~RCS relief valve~~ or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event. 3 2 3 1

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core

BASES

BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ~~[HPI or]~~ charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions. 2

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, ~~or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve,~~ or a depressurized RCS and an RCS vent of sufficient size. Two PORVs ~~RCS relief valves~~ are required for redundancy. One PORV ~~RCS relief valve~~ has adequate relieving capability to keep from overpressurization for the required coolant input capability. 3 1

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

BASES

BACKGROUND (continued)

~~[RHR Suction Relief Valve Requirements~~

~~During LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open and the RHR suction valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.~~

~~The RHR suction isolation valves and the RHR suction valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.]~~ 2

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires INSERT 1 ~~removing a pressurizer safety valve,~~ removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open. 1

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3 ~~4~~) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3~~, and in MODE 4 with RCS cold leg temperature exceeding [275°F] [LTOP arming temperature specified in the PTLR],~~ the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~ and below, overpressure prevention falls to two OPERABLE PORVs ~~RCS relief valves~~ or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability. 1 1 6 2 1

1 **INSERT 1**

an RCS vent opening of at least three square inches. This may be accomplished by removing a pressurizer safety valve,

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV ~~RCS relief valve~~ method or the depressurized and vented RCS condition. 1

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 3 ~~4~~ analyses to determine the impact of the change on the LTOP acceptance limits. 1

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters,

b. Loss of RHR cooling, or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all safety injection pumps and all but ~~[one] [HPI] pump [and~~ one charging pump~~]~~ incapable of injection, 3

b. Deactivating the accumulator discharge isolation valves in their closed positions, and

c. Disallowing start of an RCP INSERT 2 ~~if secondary temperature is more than [50]°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection~~. 1

1 **INSERT 2**

unless a steam bubble exists in the pressurizer or the secondary side water temperature of each SG is ≤ 25°F above each of the RCS cold leg temperatures. LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," provides this protection.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The Reference 3 ~~4~~ analyses demonstrate that either one PORV ~~RCS relief valve~~ 1
or the depressurized RCS and RCS vent can maintain RCS pressure
below limits when only ~~one [HPI] pump [and~~ one charging pump ~~are]~~ is
~~[are]~~ actuated. Thus, the LCO allows only ~~[one] [HPI] pump [and~~ one 2
charging pump~~]~~ OPERABLE during the LTOP MODES. Since neither
one PORV ~~RCS relief valve~~ nor the RCS vent can handle the pressure transient 1
need from accumulator injection, when RCS temperature is low, the LCO
also requires the accumulators' isolation when accumulator pressure is 6
greater than or equal to the maximum RCS pressure for the existing RCS
cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and
the valve power supply breakers fixed in their open positions. ~~The~~
~~analyses show the effect of accumulator discharge is over a narrower~~ 1
~~RCS temperature range ([175]°F and below) than that of the~~
~~LCO ([275]°F and below).~~

Fracture mechanics analyses established the temperature of LTOP
Applicability at the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~. 2 6

The consequences of a small break loss of coolant accident (LOCA) in
LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K
(Refs. 4 ~~5~~ and 5 ~~6~~), requirements by having a maximum of ~~[one] [HPI] pump~~
~~[and~~ one charging pump~~]~~ OPERABLE and SI actuation available ~~enabled~~. 2 1

PORV Performance

The fracture mechanics analyses show that the vessel is protected when
the PORVs are set to open at or below the limit shown in the PTLR. The
setpoints are derived by analyses that model the performance of the
LTOP System, assuming the limiting LTOP transient of ~~[one] [HPI] pump~~ 2
~~[and~~ one charging pump~~]~~ injecting into the RCS. These analyses
consider pressure overshoot and undershoot beyond the PORV opening
and closing, resulting from signal processing and valve stroke times. The
PORV setpoints at or below the derived limit ensures the Reference 1 P/T
limits will be met.

The PORV setpoints in the PTLR will be updated when the revised P/T
limits conflict with the LTOP analysis limits. The P/T limits are periodically
modified as the reactor vessel material toughness decreases due to
neutron embrittlement caused by neutron irradiation. Revised limits are
determined using neutron fluence projections and the results of

BASES

APPLICABLE SAFETY ANALYSES (continued)

examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

~~[RHR Suction Relief Valve Performance~~

~~The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] psig and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.~~

~~Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.~~

~~The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.]~~ 2

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 3.0 ~~2.07~~ square inches is capable of mitigating the allowed LTOP overpressure transient. 1
The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, ~~[one] HPI pump [and~~ one charging pump~~]~~ OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. 2

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

no safety injection pumps and

To limit the coolant input capability, the LCO requires that a maximum of ~~[one] [HPI] pump [and~~ one charging pump~~]~~ be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR). 3 2

three

The LCO is modified by ~~two~~ Notes. Note 1 allows ~~[~~two charging pumps~~]~~ to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than ~~[~~one~~]~~ charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. INSERT 3 2 3 2 3

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs,

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

3 **INSERT 3**

Note 3 allows a 4 hour maximum time period for rendering both safety injection and one centrifugal charging pump inoperable after entry in MODE 4 from MODE 3. RCS temperature must remain above 325°F until the pumps are rendered incapable of inadvertent injection. The 4 hour time period is sufficient for completing this activity and is based on low probability for inadvertent pump start.

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BASES

LCO (continued)

~~[b. Two OPERABLE RHR suction relief valves,~~

~~An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.~~

~~c. One OPERABLE PORV and one OPERABLE RHR suction relief valve, or]~~ 3

b ~~d~~. A depressurized RCS and an RCS vent. 3

An RCS vent is OPERABLE when open with an area of ≥ 3.0 ~~[2.07]~~ square inches. 2

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ ~~[275°F] [~~the LTOP arming temperature specified in the PTLR~~]~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above ~~[275°F] [~~the LTOP arming temperature specified in the PTLR~~]~~. When the reactor vessel head is off, overpressurization cannot occur. 2 6

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3~~, and MODE 4 above [275°F] [LTOP arming temperature specified in the PTLR]~~. 2 3

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

BASES

ACTIONS (continued)

A.1 and ~~[~~B.1~~]~~ 2 3 6

With ~~two or more HPI~~ any safety injection pump or more than one charging pump~~s~~ capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, an accumulator pressure of ~~[~~600~~]~~ psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection. 2 6

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, with one required ~~RCS relief valve~~ PORV inoperable, the ~~RCS relief valve~~ PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two ~~RCS relief valves [in any combination of the PORVS and the RHR suction relief valves]~~ PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component. 6 2 1 2

The Completion Time considers the facts that only one of the ~~RCS relief valves~~ PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. 1

BASES

ACTIONS (continued)

F.1

6

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. ~~7~~). Thus, with one of the two PORVs ~~RCS relief valves~~ inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours. 1

The Completion Time represents a reasonable time to investigate and repair several types of PORV ~~relief valve~~ failures without exposure to a lengthy period with only one OPERABLE ~~RCS relief valve~~ PORV to protect against overpressure events. 1

G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required PORVs ~~RCS relief valves~~ are inoperable, 1

b. A Required Action and associated Completion Time of Condition A, ~~[~~B~~]~~, D, E, or F is not met, or 2

c. The LTOP System is inoperable for any reason other than Condition A, ~~[~~B~~]~~, C, D, E, or F. 2

The vent must be sized ≥ 3.0 ~~[2.07]~~ square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel. 2

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, ~~[~~SR 3.4.12.2~~]~~, and SR 3.4.12.3 2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, no safety injection pumps and a maximum of ~~[one] [HPI] pump [and a maximum of~~ one charging pump~~]~~ are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out. 3 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SI

The ~~[HPI]~~ pump~~[~~s~~]~~ and charging pump~~[s]~~ are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in ~~[~~pull to lock~~]~~ and at least one valve in the discharge flow path being closed. 2 2

~~[The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT 4

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

~~[SR 3.4.12.4~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~ 3

~~The RHR suction valve is verified to be opened. [The Frequency of 12 hours is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.~~

~~OR~~

3 **INSERT 4**

The additional frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

~~The ASME Code (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.]~~ 3

4

SR 3.4.12.~~5~~ 3

3.0

The RCS vent of ≥ ~~[2.07]~~ square inches is proven OPERABLE by verifying its open condition ~~[either:~~ 2

~~a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or~~

~~b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category~~. 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

BASES

SURVEILLANCE REQUIREMENTS (continued)

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12~~d~~b. 3

SR 3.4.12.~~6~~5 3

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. ~~[~~This Surveillance is performed if the PORV satisfies the LCO.~~]~~ 2

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

~~[The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~-----]~~

~~[SR 3.4.12.7~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.]~~ 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. [The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 3

6

SR 3.4.12.~~8~~ 3

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ ~~[275°F] [~~the LTOP arming temperature specified in the PTLR~~] and every 31 days~~ on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required. 6 2 4

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time. ~~[The 31 day Frequency considers experience with equipment reliability.~~

~~OR~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275°F] [~~ the LTOP arming temperature specified in the PTLR~~]~~. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES. 2 6

7

SR 3.4.12.~~9~~ 3

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every [18] months~~ to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. ~~[The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~ 4

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.

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BASES

REFERENCES (continued)

~~3. ASME, Boiler and Pressure Vessel Code, Section III.~~

UFSAR

SII

3 ~~4~~. ~~FSAR~~, Chapter ~~[~~15~~]~~.

4 ~~5~~. 10 CFR 50, Section 50.46.

5 ~~6~~. 10 CFR 50, Appendix K.

6 ~~7~~. Generic Letter 90-06.

~~8. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

2

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but ~~[one] [high pressure injection (HPI)] pump [and~~ one centrifugal charging pump~~]~~ incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs ~~RCS relief valves~~ or a depressurized RCS and an RCS vent of sufficient size. One PORV ~~RCS relief valve~~ or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event. 3 2 3 1

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core

BASES

BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ~~[HPI or]~~ charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions. 2

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, ~~or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve,~~ or a depressurized RCS and an RCS vent of sufficient size. Two PORVs ~~RCS relief valves~~ are required for redundancy. One PORV ~~RCS relief valve~~ has adequate relieving capability to keep from overpressurization for the required coolant input capability. 3 1

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

BASES

BACKGROUND (continued)

~~[RHR Suction Relief Valve Requirements~~

~~During LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open and the RHR suction valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.~~

~~The RHR suction isolation valves and the RHR suction valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.]~~ 2

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires INSERT 1 ~~removing a pressurizer safety valve,~~ removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open. 1

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3 ~~4~~) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3~~, and in MODE 4 with RCS cold leg temperature exceeding [275°F] [LTOP arming temperature specified in the PTLR],~~ the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about ~~[275°F] [~~ the LTOP arming temperature specified in the PTLR~~]~~ and below, overpressure prevention falls to two OPERABLE PORVs ~~RCS relief valves~~ or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability. 1 1 6 2 1

1 **INSERT 1**

an RCS vent opening of at least three square inches. This may be accomplished by removing a pressurizer safety valve,

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV ~~RCS relief valve~~ method or the depressurized and vented RCS condition. 1

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 3 ~~4~~ analyses to determine the impact of the change on the LTOP acceptance limits. 1

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters,

b. Loss of RHR cooling, or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all safety injection pumps and all but ~~[one] [HPI] pump [and~~ one charging pump~~]~~ incapable of injection, 3

b. Deactivating the accumulator discharge isolation valves in their closed positions, and

c. Disallowing start of an RCP INSERT 2 ~~if secondary temperature is more than [50]°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection~~. 1

1 **INSERT 2**

unless a steam bubble exists in the pressurizer or the secondary side water temperature of each SG is ≤ 25°F above each of the RCS cold leg temperatures. LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," provides this protection.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

3 PORV

The Reference ~~4~~ analyses demonstrate that either one ~~RCS relief valve~~ or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only ~~one [HPI] pump [and~~ one charging pump ~~are]~~ is ~~[are]~~ actuated. Thus, the LCO allows only ~~[one] [HPI] pump [and~~ one charging pump~~]~~ OPERABLE during the LTOP MODES. Since neither one ~~RCS relief valve~~ nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. 1 2 1 6

PORV '

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. ~~The analyses show the effect of accumulator discharge is over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([275]°F and below).~~ 1

Fracture mechanics analyses established the temperature of LTOP Applicability at ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~. 2 6

the

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. ~~5~~ and ~~6~~), requirements by having a maximum of ~~[one] [HPI] pump [and~~ one charging pump~~]~~ OPERABLE and SI actuation ~~enabled~~. 2 1

4 5 available

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of ~~[one] [HPI] pump [and~~ one charging pump~~]~~ injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met. 2

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of

BASES

APPLICABLE SAFETY ANALYSES (continued)

examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

~~[RHR Suction Relief Valve Performance~~

~~The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] psig and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.~~

~~Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.~~

~~The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.]~~ 2

RCS Vent Performance

3.0

With the RCS depressurized, analyses show a vent size of ~~2.07~~ square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, ~~[one] HPI pump [and~~ one charging pump~~]~~ OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. 1 2

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

no safety injection pumps and

To limit the coolant input capability, the LCO requires that a maximum of ~~[one] [HPI] pump [and~~ one charging pump~~]~~ be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR). 3 2

three

The LCO is modified by ~~two~~ Notes. Note 1 allows ~~[~~two charging pumps~~]~~ to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than ~~[~~one~~]~~ charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. INSERT 3 2 3 2 3

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs,

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

3 **INSERT 3**

Note 3 allows a 4 hour maximum time period for rendering both safety injection and one centrifugal charging pump inoperable after entry in MODE 4 from MODE 3. RCS temperature must remain above 325°F until the pumps are rendered incapable of inadvertent injection. The 4 hour time period is sufficient for completing this activity and is based on low probability for inadvertent pump start.

BASES

LCO (continued)

~~[b. Two OPERABLE RHR suction relief valves,~~

~~An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.~~

~~c. One OPERABLE PORV and one OPERABLE RHR suction relief valve, or]~~ 3

b ~~d~~. A depressurized RCS and an RCS vent. 3

An RCS vent is OPERABLE when open with an area of ≥ 3.0 ~~[2.07]~~ square inches. 2

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ ~~[275°F] [~~the LTOP arming temperature specified in the PTLR~~]~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above ~~[275°F] [~~the LTOP arming temperature specified in the PTLR~~]~~. When the reactor vessel head is off, overpressurization cannot occur. 2 6

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3~~, and MODE 4 above [275°F] [LTOP arming temperature specified in the PTLR]~~. 2 3

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

BASES

ACTIONS (continued)

A.1 and ~~[~~B.1~~]~~ 2

any safety injection pump or more than one charging 3

With ~~two or more HPI~~ pump~~s~~ capable of injecting into the RCS, RCS overpressurization is possible. 6

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, an accumulator pressure of ~~[~~600~~]~~ psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection. 2 6

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ the ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~, with one required PORV ~~RCS relief valve~~ inoperable, the PORV ~~RCS relief valve~~ must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs ~~RCS relief valves [in any combination of the PORVS and the RHR suction relief valves]~~ are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component. 6 2 1 2

The Completion Time considers the facts that only one of the PORVs ~~RCS relief valves~~ is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. 1

BASES

ACTIONS (continued)

F.1

6

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. ~~7~~). Thus, with one of the two PORVs ~~RCS relief valves~~ inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours. 1

The Completion Time represents a reasonable time to investigate and repair several types of PORV ~~relief valve~~ failures without exposure to a lengthy period with only one OPERABLE ~~RCS relief valve~~ PORV to protect against overpressure events. 1

G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required PORVs ~~RCS relief valves~~ are inoperable, 1

b. A Required Action and associated Completion Time of Condition A, ~~[~~B~~]~~, D, E, or F is not met, or 2

c. The LTOP System is inoperable for any reason other than Condition A, ~~[~~B~~]~~, C, D, E, or F. 2

The vent must be sized ≥ 3.0 ~~[2.07]~~ square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel. 2

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, ~~[~~SR 3.4.12.2~~]~~, and SR 3.4.12.3 2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, no safety injection pumps and a maximum of ~~[one] [HPI] pump [and a maximum of~~ one charging pump~~]~~ are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out. 3 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SI

The ~~[HPI]~~ pump~~[~~s~~]~~ and charging pump~~[s]~~ are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in ~~[~~pull to lock~~]~~ and at least one valve in the discharge flow path being closed. 2

~~[The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT 4

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5
~~-----]~~

~~[SR 3.4.12.4~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~ 3

~~The RHR suction valve is verified to be opened. [The Frequency of 12 hours is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.~~

~~OR~~

3 **INSERT 4**

The additional frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

~~The ASME Code (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.]~~ 3

4

SR 3.4.12.~~5~~ 3

3.0

The RCS vent of ≥ ~~[2.07]~~ square inches is proven OPERABLE by verifying its open condition ~~[either:~~ 2

~~a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or~~

~~b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category~~. 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

BASES

SURVEILLANCE REQUIREMENTS (continued)

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12~~d~~b. 3

5

SR 3.4.12.~~6~~ 3

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. ~~[~~This Surveillance is performed if the PORV satisfies the LCO.~~]~~ 2

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

~~[The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

~~[SR 3.4.12.7~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.]~~ 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. [The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 3

6

SR 3.4.12.~~8~~ 3

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ ~~[275°F] [~~LTOP arming temperature specified in the PTLR~~]~~ ~~and every 31 days~~ on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required. 6 2 4

the

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time. ~~[The 31 day Frequency considers experience with equipment reliability.~~

~~OR~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275°F] [~~ the LTOP arming temperature specified in the PTLR~~]~~. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES. 2 6

SR 3.4.12.~~9~~ 7 3

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every [18] months~~ to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. ~~[The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~ 4

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.

BASES

REFERENCES (continued)

~~3. ASME, Boiler and Pressure Vessel Code, Section III.~~

SII UFSAR

3 ~~4~~. ~~FSAR~~, Chapter ~~[~~15~~]~~.

4 ~~5~~. 10 CFR 50, Section 50.46.

5 ~~6~~. 10 CFR 50, Appendix K.

6 ~~7~~. Generic Letter 90-06.

~~8. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

2

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12 Bases, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made to be consistent with changes made to the Specifications

4. ISTS SR 3.4.12.1, SR 3.4.12.2, SR 3.4.12.3, SR 3.4.12.5, SR 3.4.12.6, SR 3.4.12.8 and SR 3.4.12.9 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.12.1 through SR 3.4.12.7 under the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

6. Editorial/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 13

ITS 3.4.13, RCS OPERATIONAL LEAKAGE

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

ITS

A01

ITS 3.4.13

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

LCO 3.4.13 3.4.6.2 Reactor Coolant System leakage shall be limited to:

LCO 3.4.13.a a. No PRESSURE BOUNDARY LEAKAGE,

LCO 3.4.13.b b. 1 GPM UNIDENTIFIED LEAKAGE,

LCO 3.4.13.d c. 150 gallons per day of primary-to-secondary leakage through any one steam generator, and

LCO 3.4.13.c d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

Applicability APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

ACTION B a. With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION A b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE or primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY
ACTION B within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 4.4.6.2.1 Verify Reactor Coolant System leakage is within limits by performance of a Reactor Coolant System water inventory balance ~~at least once per 72 hours~~.* in accordance with the Surveillance Frequency Control Program LA01

~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.~~ M01

SR 3.4.13.1 Note 2 The above surveillance requirement is not applicable to primary-to-secondary leakage.

SR 3.4.13.2 4.4.6.2.2 Verify primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator ~~at least once per 72 hours~~.* in accordance with the Surveillance Frequency Control Program LA01

SR 3.4.13.1 Note 1 SR 3.4.13.2 Note

* 1. Not required to be performed until 12 hours after establishment of steady state operation.

ITS

A01

ITS 3.4.13

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

LCO 3.4.13 3.4.6.2 Reactor Coolant System leakage shall be limited to:

LCO 3.4.13.a a. No PRESSURE BOUNDARY LEAKAGE,

LCO 3.4.13.b b. 1 GPM UNIDENTIFIED LEAKAGE,

LCO 3.4.13.d c. 150 gallons per day of primary-to-secondary leakage through any one steam generator,

LCO 3.4.13.c d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

Applicability APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

ACTION B a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION A b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE or primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next

ACTION B 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 4.4.6.2.1 Verify Reactor Coolant System leakage is within limits by performance of a Reactor Coolant System water inventory balance ~~at least once per 72 hours~~ in accordance with the Surveillance Frequency Control Program.* LA01

~~The provision of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.~~ M01

SR 3.4.13.1 Note 1 The above surveillance requirement is not applicable to primary-to-secondary leakage.

SR 3.4.13.2 4.4.6.2.2 Verify primary-to-secondary leakage is $\leq$150 gallons per day through any one steam generator ~~at least once per 72 hours~~ in accordance with the Surveillance Frequency Control Program.* LA01

SR 3.4.13.1 Note 1
SR 3.4.13.2 Note

* Not required to be performed until 12 hours after establishment of steady state operation.

DISCUSSION OF CHANGES
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 4.4.6.2.1 specifies requirements to determine RCS leakage by the performance of RCS water inventory balance at least once per 72 hours but not required to be performed until 12 hours after establishment of steady state operation. The Surveillance requirement is modified by not requiring the Surveillance to be met for entry into MODES 3 or 4. ITS SR 3.4.13.1 has a similar Surveillance that requires the Surveillance to be met in the applicable MODES, but is not performed until 12 hours after establishment of steady state operation. This changes the CTS by explicitly requiring the Surveillance to be met in MODES 1, 2, 3, and 4.

The purpose of CTS 4.4.6.2.1 is to ensure the RCS leakage is within LCO limits to maintain the integrity of the reactor coolant pressure boundary. CTS 4.4.6.2.1 allows entry into MODE 3 and 4 without the surveillance being met. ITS SR 3.4.13.1 requires the Surveillance to be met prior to entering MODES 1, 2, 3, and 4, however Note 1 will still allow performance of the Surveillance to be delayed until 12 hours after steady state operation has been established in MODE 1. This change is designated as more restrictive because it imposes additional requirements on testing.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.6.2.1 requires, in part, the performance of a Reactor Coolant System water inventory balance at least once per 72 hours. CTS 4.4.6.2.2 requires, in part, that the verification of primary to secondary leakage is ≤ 150 gallons per day through any one steam generator at least once per 72 hours. ITS SRs 3.4.13.1 and SR 3.4.13.2 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

3.4.6.2 LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

3.4.6.2.a a. No pressure boundary LEAKAGE~~,~~ ; 3

3.4.6.2.b b. 1 gpm unidentified LEAKAGE~~,~~ ; 3

3.4.6.2.d c. 10 gpm identified LEAKAGE~~,~~ ; and 3

3.4.6.2.c d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
ACTION a ACTION b	B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.6.2.1 4.4.6.2.1*	SR 3.4.13.1	------------------------------NOTES---------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.6.2.2 4.4.6.2.2*	SR 3.4.13.2	------------------------------NOTE----------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

3.4.6.2 LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

3.4.6.2.a a. No pressure boundary LEAKAGE~~,~~ ; 3

3.4.6.2.b b. 1 gpm unidentified LEAKAGE~~,~~ ; 3

3.4.6.2.d c. 10 gpm identified LEAKAGE~~,~~ ; and 3

3.4.6.2.c d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
ACTION a ACTION b	B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.6.2.1 4.4.6.2.1*	SR 3.4.13.1	------------------------------NOTES------------------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
4.4.6.2.2 4.4.6.2.2*	SR 3.4.13.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after establishment of steady state operation. -- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS SR 3.4.13.1 and ISTS SR 3.4.13.2 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

3. Changes have been made to correct editorial/grammatical errors.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere INSERT 1 ~~assumes that primary to secondary LEAKAGE from all steam generators (SGs) is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis~~. 1

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The U FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via ~~safety valves~~ INSERT 2 ~~and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.~~ 1

~~The SLB is more limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes the ~~entire~~ ~~[1 gpm]~~ 150 gpd primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). 1 2

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

1 INSERT 1

assumes 150 gallons per day (gpd) per steam generator (i.e., a total of 0.4 gpm).

1 INSERT 2

the atmospheric relief valve for the affected steam generator. The 0.4 gpm operational primary to secondary leakage from all four steam generators is relatively inconsequential.

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BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~[The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.~~20~~ 17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at ~~room~~ ambient temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. 5 1

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. ~~For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.~~ RPG-011 1

For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

~~[The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. U FSAR, Section ~~[15]~~ 15.4.3. 1 2
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere INSERT 1 ~~assumes that primary to secondary LEAKAGE from all steam generators (SGs) is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis~~. 1

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The U FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via ~~safety valves~~ INSERT 2 ~~and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.~~ 1

~~The SLB is more limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes the ~~entire~~ ~~[1 gpm]~~ 150 gpd primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). 1 2

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

1 **INSERT 1**

assumes 150 gallons per day (gpd) per steam generator (i.e., a total of 0.4 gpm).

1 **INSERT 2**

the atmospheric relief valve for the affected steam generator. The 0.4 gpm operational primary to secondary leakage from all four steam generators is relatively inconsequential.

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BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~[The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.~~20~~ 17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at ~~room~~ ambient temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. 5 1

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. ~~For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.~~ RPG-011 1

For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

~~[The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. U FSAR, Section ~~[15]~~ 15.4.3. 1 2
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13 Bases, RCS OPERATIONAL LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS SR 3.4.13.1 and SR 3.4.13.2 Bases provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for ITS SR 3.4.13.1 and SR 3.4.13.2 under the Surveillance Frequency Control Program.

4. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

5. Changes have been made to reflect changes made to other Specifications.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 14

ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

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

ITS A01 ITS 3.4.14

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM PRESURE ISOLATION VALVE LEAKAGE

LIMITING CONDITION FOR OPERATION

LCO 3.4.14 SR 3.4.14.1

3.4.6.3 Leakage from each Reactor Coolant System Pressure Isolation Valve, ~~specified in Table 3.4-1,~~ shall be equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure ≥ ~~2225~~ 2215 psig and ≤ 2255 psig. LA01 SII

Applicability

APPLICABILITY: MODES 1, 2, AND 3,
MODE 4, except valves in the residual heat removal system flow path when in, or during the transition to or from, the residual heat removal mode of operation.

ACTIONS:

ACTION A / ACTION B

a. With one or more flow paths with leakage from one or more Reactor Coolant System Pressure Isolation Valves greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual, deactivated automatic, or check valve* and restore the inoperable Reactor Coolant System Pressure Isolation Valve to OPERABLE status within the following 68 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTIONS NOTES 1

b. Separate entry into the above ACTION is allowed for each flow path.

ACTIONS NOTES 2

c. Entry into the applicable ACTIONS for systems made inoperable by an inoperable Reactor Coolant System Pressure Isolation Valve is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-1~~ shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit#: LA01

SR 3.4.14.1

a. ~~At least once per 18 months~~ In accordance with the Surveillance Frequency Control Program LA02

SR 3.4.14.1

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months.

SR 3.4.14.1

c. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

SR 3.4.14.1 Note 1

Not required to be performed in MODES 3 and 4.

Required Action A.1 Note

* Each valve used to satisfy ACTION a must have been verified to meet the Surveillance Requirement 4.4.6.3 and be in the reactor coolant pressure boundary.

SR 3.4.14.1 Note 2

\# Not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the Residual Heat Removal flow path when in the shutdown cooling mode of operation.

Add proposed SR 3.4.14.1 Note 3 L01

ITS

A01

ITS 3.4.14

LA01

~~TABLE 3.4-1~~

~~REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES~~

~~VALVE NUMBER~~	~~FUNCTION~~
~~63-560~~	~~Accumulator Discharge~~
~~63-561~~	~~Accumulator Discharge~~
~~63-562~~	~~Accumulator Discharge~~
~~63-563~~	~~Accumulator Discharge~~
~~63-622~~	~~Accumulator Discharge~~
~~63-623~~	~~Accumulator Discharge~~
~~63-624~~	~~Accumulator Discharge~~
~~63-625~~	~~Accumulator Discharge~~
~~63-551~~	~~Safety Injection (Cold Leg)~~
~~63-553~~	~~Safety Injection (Cold Leg)~~
~~63-557~~	~~Safety Injection (Cold Leg)~~
~~63-555~~	~~Safety Injection (Cold Leg)~~
~~63-632~~	~~Residual Heat Removal (Cold Leg)~~
~~63-633~~	~~Residual Heat Removal (Cold Leg)~~
~~63-634~~	~~Residual Heat Removal (Cold Leg)~~
~~63-635~~	~~Residual Heat Removal (Cold Leg)~~
~~63-641~~	~~Residual Heat Removal/Safety Injection (Hot Leg)~~
~~63-644~~	~~Residual Heat Removal/Safety Injection (Hot Leg)~~
~~63-558~~	~~Safety Injection (Hot Leg)~~
~~63-559~~	~~Safety Injection (Hot Leg)~~
~~63-543~~	~~Safety Injection (Hot Leg)~~
~~63-545~~	~~Safety Injection (Hot Leg)~~
~~63-547~~	~~Safety Injection (Hot Leg)~~
~~63-549~~	~~Safety Injection (Hot Leg)~~
~~63-640~~	~~Residual Heat Removal (Hot Leg)~~
~~63-643~~	~~Residual Heat Removal (Hot Leg)~~
~~FCV-74-1*~~	~~Residual Heat Removal~~
~~FCV-74-2*~~	~~Residual Heat Removal~~

SR 3.4.14.1 Note 4

* These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 This specification has been deleted.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

This table has been deleted.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

This table has been deleted.

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ITS

A01

ITS 3.4.14

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM PRESURE ISOLATION VALVE LEAKAGE

LIMITING CONDITION FOR OPERATION

LA01

LCO 3.4.14
SR 3.4.14.1

3.4.6.3 Leakage from each Reactor Coolant System Pressure Isolation Valve, ~~specified in Table 3.4-1,~~ shall be equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure ≥ ~~2225~~ 2215 psig and ≤ 2255 psig.

SII

Applicability

APPLICABILITY: MODES 1, 2, AND 3,
MODE 4, except valves in the residual heat removal system flow path when in, or during the transition to or from, the residual heat removal mode of operation.

ACTIONS:

ACTION A
ACTION B

a. With one or more flow paths with leakage from one or more Reactor Coolant System Pressure Isolation Valves greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual, deactivated automatic, or check valve* and restore the inoperable Reactor Coolant System Pressure Isolation Valve to OPERABLE status within the following 68 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTIONS NOTES 1

b. Separate entry into the above ACTION is allowed for each flow path.

ACTIONS NOTES 2

c. Entry into the applicable ACTIONS for systems made inoperable by an inoperable Reactor Coolant System Pressure Isolation Valve is required.

SURVEILLANCE REQUIREMENTS

LA01

SR 3.4.14.1

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-1~~ shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit#:

In accordance with the Surveillance Frequency Control Program

LA02

SR 3.4.14.1

a. ~~At least once per 18 months~~

SR 3.4.14.1

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months.

SR 3.4.14.1

c. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

SR 3.4.14.1 Note 1

Not required to be performed in MODES 3 and 4.

Required Action A.1 Note

* Each valve used to satisfy ACTION a must have been verified to meet the Surveillance Requirement 4.4.6.3 and be in the reactor coolant pressure boundary.

SR 3.4.14.1 Note 2

\# Not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the Residual Heat Removal flow path when in the shutdown cooling mode of operation.

Add proposed SR 3.4.14.1 Note 3

L01

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ITS

A01

ITS 3.4.14

LA01

~~TABLE 3.4-1~~

~~REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES~~

~~VALVE NUMBER~~	~~FUNCTION~~
~~63-560~~	~~Accumulator Discharge~~
~~63-561~~	~~Accumulator Discharge~~
~~63-562~~	~~Accumulator Discharge~~
~~63-563~~	~~Accumulator Discharge~~
~~63-622~~	~~Accumulator Discharge~~
~~63-623~~	~~Accumulator Discharge~~
~~63-624~~	~~Accumulator Discharge~~
~~63-625~~	~~Accumulator Discharge~~
~~63-551~~	~~Safety Injection (Cold Leg)~~
~~63-553~~	~~Safety Injection (Cold Leg)~~
~~63-557~~	~~Safety Injection (Cold Leg)~~
~~63-555~~	~~Safety Injection (Cold Leg)~~
~~63-632~~	~~Residual Heat Removal (Cold Leg)~~
~~63-633~~	~~Residual Heat Removal (Cold Leg)~~
~~63-634~~	~~Residual Heat Removal (Cold Leg)~~
~~63-635~~	~~Residual Heat Removal (Cold Leg)~~
~~63-641~~	~~Residual Heat Removal/Safety Injection (Hot Leg)~~
~~63-644~~	~~Residual Heat Removal/Safety Injection (Hot Leg)~~
~~63-558~~	~~Safety Injection (Hot Leg)~~
~~63-559~~	~~Safety Injection (Hot Leg)~~
~~63-543~~	~~Safety Injection (Hot Leg)~~
~~63-545~~	~~Safety Injection (Hot Leg)~~
~~63-547~~	~~Safety Injection (Hot Leg)~~
~~63-549~~	~~Safety Injection (Hot Leg)~~
~~63-640~~	~~Residual Heat Removal (Hot Leg)~~
~~63-643~~	~~Residual Heat Removal (Hot Leg)~~
~~FCV-74-1*~~	~~Residual Heat Removal~~
~~FCV-74-2*~~	~~Residual Heat Removal~~

SR 3.4.14.1 Note 4

* These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 This specification has been deleted.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

This table has been deleted.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

This table has been deleted.

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DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.6.3 requires the leakage from each RCS PIV specified in Table 3.4-1 to be limited and CTS 4.4.6.3 requires the RCS PIVs in Table 3.4-1 to be periodically tested. ITS 3.4.14 does not contain nor make reference to an RCS PIV Table. This changes the CTS by relocating the list of the PIVs to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of the public health and safety. The ITS still requires the RCS PIVs to be OPERABLE. It is not necessary for the list of the RCS PIVs to be in the Technical Specifications in order to ensure that the RCS PIVs are OPERABLE. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.6.3.a requires, in part, the RCS PIVs be demonstrated OPERABLE at least once per 18 months. ITS SR 3.4.14.1 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the

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DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.4.6.3 requires, in part, the performance of the RCS PIV leakage test pursuant to Specification 4.0.5. ITS SR 3.4.14.1 requires a similar Surveillance and includes a Note that provides an allowance for RCS PIVs actuated during the performance of the Surveillance. Testing of the RCS PIV is not required to be performed more than once if a repetitive testing loop cannot be avoided. This changes the CTS by providing an allowance not requiring the performance of the Surveillance more than once if a repetitive testing loop cannot be avoided.

The purpose of CTS 4.4.6.3 is to perform the RCS PIV leakage test in accordance with the frequency of the Inservice Test Program (CTS 4.0.5). The SR is modified by a Note that states if a RCS PIV is actuated during the performance of the Surveillance, testing of the RCS PIV is only required once if a repetitive testing loop cannot be avoided. Therefore, if a condition exists during the performance of the Surveillance that disturbs the PIV seat leakage test, the Surveillance must be performed within 24 hours after the valve has reseated. This change is acceptable because 24 hours is a reasonable and practical time limit to perform the Surveillance after the RCS PIV is seated. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than the CTS.

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

CTS

RCS PIV Leakage
3.4.14

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

3.4.6.3 LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

Applicability APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

------------------------------NOTES------------------------------

ACTION b 1. Separate Condition entry is allowed for each flow path.

ACTION c 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable RCS PIV. 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ACTION a* A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	----------NOTE---------- Each valve used to satisfy Required Action A.1 ~~and Required Action A.2~~ must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary ~~[or the high pressure portion of the system]~~. ------------------------------	1 2
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. AND	4 hours

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a		A.2 ~~[Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.~~ ~~[or]~~ Restore RCS PIV to within limits.	~~72 hours~~ 72 hours ~~]~~
ACTION a	B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
	~~C. [RHR System autoclosure interlock function inoperable.~~	~~C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.~~	~~4 hours]~~

2

3

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CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
3.4.6.3 4.4.6.3.a 4.4.6.3.b 4.4.6.3.c 4.4.6.3 # SR 3.4.14.1	------------------------------NOTES----------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. Table 3.4-1* 4. Not required to be performed for RCS PIVs FCV-74-1 and FCV-74-2 following manual or automatic actuation or flow through the valves. --- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ ~~[2215]~~ ~~2225~~ 2215 psig and ≤ ~~[~~2255~~]~~ psig. SII	4 In accordance with the Inservice Testing Program, ~~and [[18] months~~ stet 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 5 AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		Within 24 hours following valve actuation due to automatic or manual action or flow through the valve
~~SR 3.4.14.2~~	~~-------------------------NOTE-------------------------~~ ~~[Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ ~~--~~ ~~Verify RHR System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [425] psig.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
~~SR 3.4.14.3~~	~~-------------------------NOTE-------------------------~~ ~~[Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ ~~--~~ ~~Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [600] psig.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

3

6

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

3.4.6.3 LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

Applicability APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

--NOTES--

ACTION b 1. Separate Condition entry is allowed for each flow path.

ACTION c 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable RCS PIV. 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ACTION a* A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	--------------------NOTE------------------- Each valve used to satisfy Required Action A.1 ~~and Required Action A.2~~ must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary ~~[or the high pressure portion of the system]~~. ---	1 2
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. AND	4 hours

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a		A.2 ~~[Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.~~ ~~[or]~~ Restore RCS PIV to within limits.	~~72 hours~~ 72 hours]	2
ACTION a	B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours	
	~~C. [RHR System autoclosure interlock function inoperable.~~	~~C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.~~	~~4 hours]~~	3

CTS

RCS PIV Leakage
3.4.14

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
3.4.6.3 4.4.6.3.a 4.4.6.3.b 4.4.6.3.c 4.4.6.3 # SR 3.4.14.1	-----------------------------NOTES---------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. Table 3.4-1* 4. Not required to be performed for RCS PIVs FCV-74-1 and FCV-74-2 following manual or automatic actuation or flow through the valves. -- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ ~~[2215]~~ ~~2225~~ 2215 psig and ≤ ~~[~~2255~~]~~ psig. SII	stet In accordance with the Inservice Testing Program, ~~and [[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2 5 4 AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		Within 24 hours following valve actuation due to automatic or manual action or flow through the valve
~~SR 3.4.14.2~~	~~-----NOTE-----~~ ~~[Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ ~~----------~~ ~~Verify RHR System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [425] psig.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
~~SR 3.4.14.3~~	~~-----NOTE-----~~ ~~[Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ ~~----------~~ ~~Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [600] psig.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The requirements of ISTS 3.4.14 Condition C and SR 3.4.14.2 have been deleted consistent with the current licensing basis. Opening of the RHR suction line isolation valves is administratively controlled, as described in UFSAR Section 5.5.7.2.

3.4.14.1

4. ISTS SR ~~3.47.14.1~~ is changed with the addition of Note 4, thereby not requiring the performance of the SR for valves FCV-74-1 and FCV-74-2 following a manual or automatic actuation or flow through the valves. This change is consistent with the current licensing basis as stated in CTS Table 3.4-1 Note*. SII

5. ISTS SR 3.4.14.1 provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency under the Surveillance Frequency Control Program.

6. ISTS SR 3.4.14.3 has been deleted consistent with current licensing basis. SQN design does not include an autoclosure interlock that automatically closes the RHR System suction valves on a high RCS pressure signal.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System,

BASES

BACKGROUND (continued)

b. Safety Injection System, and

c. Chemical and Volume Control System.

Table B 3.4.14-1

The PIVs are listed in ~~the FSAR, Section [ ] (Ref. 6)~~. 1

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

BASES

LCO (continued)

6

Reference ~~7~~ permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power. 2

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB ~~[or the high pressure portion of the system]~~. 3

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

BASES

ACTIONS (continued)

~~[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.~~ 3

~~[or]~~

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete this Action and the low probability of a second valve failing during this period. ~~]~~

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.~~
~~---~~ 4

B.1 and B.2 SII 5

Required Actions and associated Completion Times of Condition A are not met

If ~~leakage cannot be reduced, [the system can not be isolated,] or the other Required Actions accomplished~~, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~C.1~~

~~The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function.~~ 5

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every ~~[~~9~~]~~ months, but may be extended, if the plant does not go into MODE 5 for at least 7 days. ~~[The [18 month] Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

stet

7

6

3

SII

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

4

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

~~[SR 3.4.14.2 and SR 3.4.14.3~~

~~Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. [The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 5

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.]~~ 5

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, Section V, GDC 55.

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

5. NUREG-0677, May 1980.

~~[6. Document containing list of PIVs.]~~ 1

6 ~~7~~. ASME Code for Operation and Maintenance of Nuclear Power Plants. 2

SII

7. ~~8.~~ ~~10 CFR 50.55a(g).~~ stet 1

INSERT 1 1

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1 **INSERT 1**

Table B 3.4.14-1 (page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
FCV-74-1	Residual Heat Removal
FCV-74-2	Residual Heat Removal

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System,

BASES

BACKGROUND (continued)

b. Safety Injection System, and

c. Chemical and Volume Control System.

Table B 3.4.14-1

The PIVs are listed in ~~the FSAR, Section [ ] (Ref. 6)~~. 1

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

BASES

LCO (continued)

6

Reference ~~7~~ permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power. 2

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB ~~[or the high pressure portion of the system]~~. 3

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

BASES

ACTIONS (continued)

~~[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.~~

~~[or]~~

3

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete this Action and the low probability of a second valve failing during this period. ~~]~~

~~-----REVIEWER'S NOTE-----~~
~~Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.~~
~~-----~~

4

B.1 and B.2

Required Actions and associated Completion Times of Condition A are not met

SII

5

If ~~leakage cannot be reduced, [the system can not be isolated,] or the other Required Actions accomplished~~, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~C.1~~

~~The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function.~~

5

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every ~~[~~9~~]~~ months, but may be extended, if the plant does not go into MODE 5 for at least 7 days. ~~[The [18 month] Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

stet

3

SII

~~6~~ 3

~~OR~~

7

6

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

4

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

~~[SR 3.4.14.2 and SR 3.4.14.3~~

~~Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. [The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 5

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.]~~ 5

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.

~~[6. Document containing list of PIVs.]~~ 1

6 ~~7~~. ASME Code for Operation and Maintenance of Nuclear Power Plants. 2

7. ~~8.~~ ~~10 CFR 50.55a(g).~~ stet 1 SII

INSERT 1 1

1 **INSERT 1**

Table B 3.4.14-1 (page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
FCV-74-1	Residual Heat Removal
FCV-74-2	Residual Heat Removal

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14 BASES, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The renumbering of references has been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 2.7.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The Bases are changed to reflect changes made to the Specification.

6. ISTS SR 3.4.14.1 provides two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.4.14.1 is "In accordance with the Surveillance Frequency Control Program."

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 15

ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

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

ITS

A01

ITS 3.4.15

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.4.15

3.4.6.1 The following Reactor Coolant System leakage detection instrumentation shall be OPERABLE:

LCO 3.4.15.b

a. One lower containment atmosphere particulate radioactivity monitoring channel, and

LCO 3.4.15.a

b. One containment pocket sump level monitor.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A
ACTION C

a. With both containment pocket sump monitors inoperable, operation may continue for up to 30 days provided SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION B
ACTION C

b. With the particulate lower containment atmosphere radioactivity monitor inoperable, operation may continue for up to 30 days provided grab samples of the lower containment atmosphere are analyzed once per 24 hours or SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION D

c. With both containment pocket sump monitors and the lower containment atmosphere radioactivity monitor inoperable, ~~be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

Add proposed Required Action D.1 and associated Completion Time

L01

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

OPERATIONAL

M01

SR 3.4.15.1
SR 3.4.15.2
SR 3.4.15.4

a. Performance of the lower containment atmosphere particulate monitor CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL ~~FUNCTIONAL~~ TEST ~~at the frequencies specified in Table 4.3-3, and~~

in accordance with the Surveillance Frequency Control Program

LA01

SR 3.4.15.3

b. Performance of containment pocket sump level monitor CHANNEL CALIBRATION ~~at least once per 18 months~~.

in accordance with the Surveillance Frequency Control Program

LA01

Required Action A.1 and B.1.2 Note

* Surveillance performance not required until 12 hours after establishment of steady state operation.

CTS A01 ITS 3.4.15

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
1. AREA MONITOR						
a. Fuel Storage Pool Area	1	*	≤ 151 mR/hr	10^{-1} - 10^{4} mR/hr	26	See ITS 3.3.8
2. PROCESS MONITORS						
a. Containment Purge Air	1	1, 2, 3, 4 & 6	≤ 8.5x 10^{-3} μCi/cc	10 - 10^{7} cpm	28	See ITS 3.3.6
b. Containment						
i. Deleted						
LCO 3.4.15.b ii. Particulate Activity					ACTION B	
RCS Leakage Detection	1	1, 2, 3 & 4	N/A	~~10 - 10^{7} cpm~~	$\boxed{27}$	LA02
c. Control Room Isolation	2	ALL MODES and during movement of irradiated fuel assemblies	≤ 400 cpm**	10 - 10^{7} cpm	29	See ITS 3.3.7

* With fuel in the storage pool or building — See ITS 3.3.8

** Equivalent to 1.0 x 10^{-5} μCi/cc. — See ITS 3.3.7

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 26 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

LCO 3.4.15 ACTION B

ACTION 27 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 28 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9 (MODE 6) and 3.3.2.1 (MODES 1, 2, 3, and 4).

See ITS 3.3.6

ACTION 29 -

a. With one channel inoperable, place the associated control room emergency ventilation system (CREVS) train in recirculation mode of operation within 7 days or be at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two channels inoperable, within 1 hour initiate and maintain operation of one CREVS train in the recirculation mode of operation and enter the required Actions for one CREVS train made inoperable by inoperable CREVS actuation instrumentation.

Or

place both trains in the recirculation mode of operation within one hour.

If the completion time of Action 29b cannot be met in Modes 1, 2, 3, and 4, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If the completion time of Action 29b cannot be met during the movement of irradiated fuel assemblies, suspend core alterations and suspend movement of irradiated fuel assemblies.

If the completion time of Action 29b cannot be met in Modes 5 and 6, initiate action to restore one CREVS train.

See ITS 3.3.7

CTS A01 ITS 3.4.15

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	SR 3.4.15.1 CHANNEL CHECK	SR 3.4.15.4 CHANNEL CALIBRATION	SR 3.4.15.2 CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE REQUIRED	
1. AREA MONITOR					M01
a. Fuel Storage Pool Area	S	R	Q	*	See ITS 3.3.8
2. PROCESS MONITORS					
a. Containment Purge Air Exhaust	S	R	Q	1, 2, 3, 4 & 6	See ITS 3.3.6
b. Containment					
i. Deleted					
ii. Particulate Activity	In accordance with the Surveillance Frequency Control Program				LA01
RCS Leakage Detection	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4	
c. Control Room Isolation	S	R	Q	ALL MODES	See ITS 3.3.7

*With fuel in the storage pool or building. See ITS 3.3.8

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 This specification has been deleted.

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INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 This specification is deleted.

(Pages 3/4 3-44 through 3/4 3-46 are deleted)

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INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 This specification has been deleted.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

This table has been deleted.

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

This table has been deleted.

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.4.15 3.4.6.1 The following Reactor Coolant System leakage detection instrumentation shall be OPERABLE:

LCO 3.4.15.b a. One lower containment atmosphere particulate radioactivity monitoring channel, and

LCO 3.4.15.a b. One containment pocket sump level monitor.

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

ACTION:

ACTION A
ACTION C
a. With both containment pocket sump monitors inoperable, operation may continue for up to 30 days provided SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION B
ACTION C
b. With the particulate lower containment atmosphere radioactivity monitor inoperable, operation may continue for up to 30 days provided grab samples of the lower containment atmosphere are analyzed once per 24 hours or SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION D
c. With both containment pocket sump monitors and the lower containment atmosphere radioactivity monitor inoperable, ~~be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

Add proposed Required Action D.1 and associated Completion Time L01

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

SR 3.4.15.1
SR 3.4.15.2
SR 3.4.15.4
a. Performance of the lower containment atmosphere particulate monitor CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST ~~at the frequencies specified in Table 4.3-3, and~~ in accordance with the Surveillance Frequency Control Program

M01 LA01

SR 3.4.15.3
b. Performance of containment pocket sump level monitor CHANNEL CALIBRATION ~~at least once per 18 months~~ in accordance with the Surveillance Frequency Control Program.

LA01

Required Action A.1 and B.1.2 Note
* Surveillance performance not required until 12 hours after establishment of steady state operation.

ITS A01 ITS 3.4.15

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
1. AREA MONITOR						
a. Fuel Storage Pool Area	1	*	≤151 mR/hr	10^{-1} - 10^{4} mR/hr	26	See ITS 3.3.8
2. PROCESS MONITORS						
a. Containment Purge Air	1	1, 2, 3, 4 & 6	≤8.5 x 10^{-3} µCi/cc	10 - 10^{7} cpm	28	See ITS 3.3.6
b. Containment						
i. Deleted						
LCO 3.4.15.b ii. Particulate Activity					ACTION B	
RCS Leakage Detection	1	1, 2, 3 & 4	N/A	~~10 - 10^{7} cpm~~	$\boxed{27}$	LA02
c. Control Room Isolation	2	ALL MODES and during movement of irradiated fuel assemblies	≤ 400 cpm**	10 - 10^{7} cpm	29	See ITS 3.3.7

* With fuel in the storage pool or building — See ITS 3.3.8

** Equivalent to 1.0 x 10^{-5} µCi/cc. — See ITS 3.3.7

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TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 26 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

LCO 3.4.15 ACTION B

ACTION 27 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 28 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9 (MODE 6) and 3.3.2 (MODES 1, 2, 3, and 4).

See ITS 3.3.6

ACTION 29 -

a. With one channel inoperable, place the associated control room emergency ventilation system (CREVS) train in recirculation mode of operation within 7 days or be at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two channels inoperable, within 1 hour initiate and maintain operation of one CREVS train in the recirculation mode of operation and enter the required Actions for one CREVS train made inoperable by inoperable CREVS actuation instrumentation.

Or

place both trains in the recirculation mode of operation within one hour.

If the completion time of Action 29b cannot be met in Modes 1, 2, 3, and 4, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If the completion time of Action 29b cannot be met during the movement of irradiated fuel assemblies, suspend core alterations and suspend movement of irradiated fuel assemblies.

If the completion time of Action 29b cannot be met in Modes 5 and 6, initiate action to restore one CREVS train.

See ITS 3.3.7

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ITS A01 ITS 3.4.15

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	SR 3.4.15.1 CHANNEL CHECK	SR 3.4.15.4 CHANNEL CALIBRATION	SR 3.4.15.2 CHANNEL ~~FUNCTIONAL~~ TEST	OPERATIONAL MODES FOR WHICH SURVEILLANCE IS REQUIRED	M01
1. AREA MONITOR					
a. Fuel Storage Pool Area	S	R	Q	*	See ITS 3.3.8
2. PROCESS MONITORS					
a. Containment Purge Air Exhaust	S	R	Q	1, 2, 3, 4 & 6	See ITS 3.3.6
b. Containment					
i. Deleted					
ii. Particulate Activity RCS Leakage Detection	~~S~~ In accordance with the Surveillance Frequency Control Program	~~R~~	~~Q~~	1, 2, 3 & 4	LA01
c. Control Room Isolation	S	R	Q	ALL MODES	See ITS 3.3.7

* With fuel in the storage pool or building. See ITS 3.3.8

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 This specification has been deleted.

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INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 This specification is deleted

(Pages 3/4 3-45 through 3/4 3-47 are deleted)

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INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 This specification has been deleted.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

This table has been deleted.

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

This table has been deleted.

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DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 4.4.6.1 requires, in part, the lower containment atmosphere radioactivity monitor to be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the Frequencies shown in Table 4.3-3. ITS SR 3.4.15.2 requires the performance of a CHANNEL OPERATIONAL TEST (COT). This changes the CTS by changing the CHANNEL FUNCTIONAL TEST requirements to a COT.

CTS defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated signal into the sensor as close to the sensor as practicable to verify OPERABILITY. ITS defines a COT as the injection of an actual or simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. This changes the CTS by requiring adjustments of the setpoint so that the lower containment atmosphere radioactivity monitor is within the necessary range and accuracy. This change is designated as more restrictive because it imposes additional requirements on testing.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.4.6.1.a requires, in part, the containment atmosphere particulate monitor be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-3. CTS 4.4.6.1.b requires, in part, that the containment pocket sump level monitor be demonstrated OPERABLE by the performance of a CHANNEL CALIBRATION at least once per 18 months. ITS SRs 3.4.15.1, SR 3.4.15.2, SR 3.4.15.3, and SR 3.4.15.4 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 provides the measurement range for the particulate containment atmosphere radioactivity monitor. ITS 3.4.15 requires the particulate monitor to be OPERABLE, but the details concerning the measurement range are not included. This changes the CTS by moving the details of the measurement range for the particulate containment atmosphere radioactivity monitor to the UFSAR.

The removal of these details from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of the public health and safety. The ITS still retains the requirement for an OPERABLE particulate containment radioactivity monitor and Surveillances to ensure the particulate containment atmosphere radioactivity monitor is capable of performing its safety function. This change is acceptable because the UFSAR contains the design measurement range of particulate containment atmosphere radioactivity monitor. The removed requirements will be adequately controlled in the UFSAR as any changes to the UFSAR are made under 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal

DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

of detail change, because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.4.6.1 ACTION c requires when the containment pocket sump monitor and the lower containment atmosphere radioactivity monitor are inoperable to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.4.15 ACTION D requires, for the same type of inoperability, to enter LCO 3.0.3 immediately. ITS LCO 3.0.3 requires ACTION to be initiated within one hour to place the unit in MODE 3 within 7 hours, MODE 4 within 13 hours, and MODE 5 within 37 hours. This changes the CTS by requiring entry into LCO 3.0.3 when the containment pocket sump monitor and the lower containment atmosphere radioactivity monitor are inoperable.

With the containment pocket sump monitor and the lower containment atmosphere radioactivity monitor inoperable, no automatic means of monitoring leakage is available. The convention of the ITS is to require an immediate entry into LCO 3.0.3 when the intended function cannot be met. This change is designated as less restrictive because additional time is allowed to bring the unit to MODE 3 and MODE 5.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

3.4.6.1
Table 3.3-6

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

3.4.6.1.b

a. One containment (pocket) sump ~~(~~level ~~or discharge flow)~~ monitor~~,~~ (; and) 1

3.4.6.1.a
Table 3.3-6 Instrument 2.b.ii

b. One (lower) containment atmosphere (particulate) radioactivity monitor ~~(gaseous or particulate), and~~ 1

~~[c. One containment air cooler condensate flow rate monitor.]~~ 2

3.4.6.1.a
Applicability
Table 3.3-6
Instrument 2.b.ii
Applicability

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment (pocket) sump (level) monitor inoperable.	A.1 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.2 Restore required containment (pocket) sump (level) monitor to OPERABLE status.	30 days

3.4.6.1 ACTION a
3.4.6.1 ACTION a*

1

1

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
3.4.6.1 ACTION b 3.4.6.1 ACTION b* Table 3.3-6 Instrument 2.b.ii Action 27 B. Required lower containment atmosphere particulate radioactivity monitor inoperable.	B.1.1	Analyze grab samples of the lower containment atmosphere.	Once per 24 hours
	OR		
	B.1.2	--------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	B.2.~~1~~	Restore required lower containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
	~~OR~~		
	~~[B.2.2~~	~~Verify containment air cooler condensate flow rate monitor is OPERABLE.~~	~~30 days~~
~~C. [Containment air cooler condensate flow rate monitor inoperable.~~	~~C.1~~	~~Perform SR 3.4.15.1.~~	~~Once per 8 hours~~
	~~OR~~		
	~~C.2~~	~~--------------NOTE--------------~~ ~~Not required until 12 hours after establishment of steady state operation.~~ ~~-------------------------------------~~ ~~Perform SR 3.4.13.1.~~	~~Once per 24 hours]~~

1

2 1 1

2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~------------ NOTE ------------~~ ~~Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor.~~ ~~----------------------------------~~ ~~D. Required containment sump monitor inoperable.~~ ~~AND~~ ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~D.1 Analyze grab samples of the containment atmosphere.~~ ~~AND~~ ~~D.2.1 Restore required containment sump monitor to OPERABLE status.~~ ~~OR~~ ~~[D.2.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	~~Once per 12 hours~~ ~~7 days~~ ~~7 days~~
~~E. [Required containment atmosphere radioactivity monitor inoperable.~~ ~~AND~~ ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~E.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status.~~ ~~OR~~ ~~[E.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	~~30 days~~ ~~30 days]~~
C ~~F~~. Required Action and associated Completion Time of Condition A or B not met.	C ~~F~~.1 Be in MODE 3. AND C ~~F~~.2 Be in MODE 5.	6 hours 36 hours
D ~~G~~. All required monitors inoperable.	D ~~G~~.1 Enter LCO 3.0.3.	Immediately

3.4.6.1 ACTION a
3.4.6.1 ACTION b
Table 3.3-6 Instrument 2.b.ii Action 27
3.4.6.1 ACTION c

2
2
3
2
2

1

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.1	Perform CHANNEL CHECK of the required lower containment atmosphere particulate radioactivity monitor.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 1 4
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.2	Perform COT of the required lower containment atmosphere particulate radioactivity monitor.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 1 4

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.6.1.b	SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment pocket sump level monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 1 3
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.4	~~[~~ Perform CHANNEL CALIBRATION of the required lower containment atmosphere particulate radioactivity monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	3 1 3
	~~SR 3.4.15.5~~	~~[Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

3.4.6.1
Table 3.3-6

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

3.4.6.1.b

a. One containment pocket sump ~~(~~level ~~or discharge flow)~~ monitor~~.~~ ; and

3.4.6.1.a
Table 3.3-6 Instrument 2.b.ii

b. One lower containment atmosphere particulate radioactivity monitor ~~(gaseous or particulate), and~~ 1

~~[c. One containment air cooler condensate flow rate monitor.]~~ 2

3.4.6.1.a
Applicability
Table 3.3-6
Instrument 2.b.ii
Applicability

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment pocket sump level monitor inoperable.	A.1 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.2 Restore required containment pocket sump level monitor to OPERABLE status.	30 days

3.4.6.1
ACTION a
3.4.6.1
ACTION a*

1

1

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
3.4.6.1 ACTION b; 3.4.6.1 ACTION b*; Table 3.3-6 Instrument 2.b.ii Action 27 — B. Required lower containment atmosphere particulate radioactivity monitor inoperable.	B.1.1	Analyze grab samples of the lower containment atmosphere.	Once per 24 hours
	OR		
	B.1.2	--------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------ Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	B.2~~.1~~	Restore required lower containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
	~~OR~~		
	~~[B.2.2~~	~~Verify containment air cooler condensate flow rate monitor is OPERABLE.~~	~~30 days~~
~~C. [Containment air cooler condensate flow rate monitor inoperable.~~	~~C.1~~	~~Perform SR 3.4.15.1.~~	~~Once per 8 hours~~
	~~OR~~		
	~~C.2~~	~~--------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------ Perform SR 3.4.13.1.~~	~~Once per 24 hours]~~

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~------------ NOTE ------------~~ ~~Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor.~~ ~~------------------------------~~ ~~D. Required containment sump monitor inoperable.~~ ~~AND~~ ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~D.1 Analyze grab samples of the containment atmosphere.~~ ~~AND~~ ~~D.2.1 Restore required containment sump monitor to OPERABLE status.~~ ~~OR~~ ~~[D.2.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	~~Once per 12 hours~~ ~~7 days~~ ~~7 days~~
~~E. [Required containment atmosphere radioactivity monitor inoperable.~~ ~~AND~~ ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~E.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status.~~ ~~OR~~ ~~[E.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	~~30 days~~ ~~30 days]~~
C ~~F~~. Required Action and associated Completion Time of Condition A or B not met.	C ~~F~~.1 Be in MODE 3. AND C ~~F~~.2 Be in MODE 5.	6 hours 36 hours
D ~~G~~. All required monitors inoperable.	D ~~G~~.1 Enter LCO 3.0.3.	Immediately

3.4.6.1 ACTION a
3.4.6.1 ACTION b
Table 3.3-6 Instrument 2.b.ii Action 27
3.4.6.1 ACTION c

2
2
3
2
2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.1 Perform CHANNEL CHECK of the required lower containment atmosphere particulate radioactivity monitor.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 1 4
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.2 Perform COT of the required lower containment atmosphere particulate radioactivity monitor.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 4 1 4

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SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
4.4.6.1.b	SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment [pocket] sump [level] monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 1 3
4.4.6.1.a Table 4.3-3 Instrument 2.b.ii	SR 3.4.15.4	~~[~~ Perform CHANNEL CALIBRATION of the required [lower] containment atmosphere [particulate] radioactivity monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	3 1 3
	~~SR 3.4.15.5~~	~~[Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS LCO 3.4.15.c is a bracketed requirement and states that one containment air cooler condensate flow rate monitor is required to be OPERABLE. SQN does not have a containment air cooler condensate flow rate monitor. CTS 3.4.6.1 only requires two types of leakage detection instrumentation; pocket sump monitoring and containment atmosphere particulate radioactivity monitor. Therefore, the bracketed ISTS LCO 3.4.15.c requirement has not been included in the ITS. Due to this deletion, ISTS 3.4.15 ACTIONS C and D and optional Required Action B.2.2 and SR 3.4.15.5 have been deleted, since they apply to the containment air cooler condensate flow rate monitor. Subsequent ACTIONS have been renumbered.

3. The specific Condition the ACTION applies to has been added, since there is one ACTION it does not apply to (ISTS 3.4.15 ACTION F, ITS 3.4.15 ACTION D). This is consistent with the writers guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 4.1.6.i. 5.ii.

4. ISTS SR 3.4.15.1, ISTS SR 3.4.15.2, ISTS SR 3.4.15.3, and ISTS SR 3.4.15.4 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. ~~[In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms.]~~ 1

The containment pocket sump used to collect unidentified LEAKAGE ~~[~~is~~]~~ ~~[(or) and the containment air cooler condensate flow rate monitor] [are]~~ instrumented to alarm for increases above the normal flow rates. 2 1

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. A r~~R~~adioactivity detection system~~s~~ ~~are~~ is included for monitoring ~~both~~ particulate activity ~~and gaseous activities~~ because of ~~their sensitivities~~ its sensitivity and rapid response~~s~~ to RCS LEAKAGE. 2

Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment pocket sump ~~[and condensate flow from air coolers]~~. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO. 2 1

BASES

BACKGROUND (continued)

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. ~~[~~Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.~~]~~

1

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. ~~[~~Some of these systems could ~~service~~ serve as early alarm systems signaling the ~~operations~~ operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.~~]~~ 6 6

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires ~~[three]~~ two instruments to be OPERABLE.

The containment pocket sump is used to collect unidentified LEAKAGE. ~~[The containment sump consists of the normal sump and the emergency sump. The LCO requirements apply to the total amount of unidentified LEAKAGE collected in [the][both] sump[s].]~~ The monitor on the containment pocket sump detects ~~[~~level ~~or flow rate or the operating frequency of a pump]~~ and is instrumented to detect when there is ~~[leakage of]~~ ~~[~~an 2 1 2

BASES

LCO (continued)

increase above the normal value by~~]~~ 1 gpm. The identification of ~~[~~an increase in~~]~~ unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment pocket sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment pocket sump level monitor OPERABILITY. 1 2 2

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the ~~gaseous or particulate~~ lower containment atmosphere particulate radioactivity monitor. ~~Only one of the two detectors is required to be OPERABLE.~~ A r~~R~~adioactivity detection system~~s~~ ~~are~~ is included for monitoring ~~both~~ particulate activity ~~and gaseous activities~~ because of the~~ir~~ sensitivity ~~sensitivities~~ and rapid response~~s~~ to RCS LEAKAGE, but ha~~ve~~ as recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the ~~gaseous or~~ lower ~~particulate~~ containment atmosphere particulate radioactivity monitor~~s~~ to detect a 1 gpm increase within 1 hour during normal operation. However, the ~~gaseous or~~ lower ~~particulate~~ containment atmosphere particulate radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitor~~s~~ (Reference 3). 2

~~[An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.]~~ 1

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment pocket sump level monitor, in combination with a ~~gaseous or~~ particulate radioactivity monitor ~~[and a containment air cooler condensate flow rate monitor]~~, provides an acceptable minimum. 2 1

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

pocket level

With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information; however, the particulate lower containment atmosphere radioactivity monitor will provide indications of lower changes in leakage. Together with the containment atmosphere particulate radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2 1

pocket level

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1. 2

SII

B.1.1, B.1.2, and B.2

~~B.1.1, B.1.2, B.2.1, and B.2.2~~ 3

the lower particulate

With ~~both gaseous and particulate~~ containment atmosphere radioactivity monitoring instrumentation channel~~s~~ inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. 2

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days lower particulate to allow restoration of the required containment atmosphere radioactivity monitors. ~~[Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances performed every 24 hours.]~~ 1

BASES

ACTIONS (continued)

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available. 1

~~[C.1 and C.2~~

~~With the containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.~~ 3

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.]~~

~~D.1, D.2.1, and D.2.2~~

~~With the required containment sump monitor [and the containment air cooler condensate flow rate monitor] inoperable, the only means of detecting LEAKAGE is the required containment atmosphere radiation monitor. A Note clarifies that this Condition is only applicable when the only OPERABLE monitor is the required containment atmosphere gaseous radiation monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the containment atmosphere must be taken to provide alternate periodic information. The 12 hour interval is sufficient to detect increasing RCS~~ 3

BASES

ACTIONS (continued)

~~leakage. The Required Action provides 7 days to restore another RCS leakage monitor to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.~~ 3

~~[E.1 and E.2~~

~~With the required containment atmosphere radioactivity monitor [and the containment air cooler condensate flow rate monitor] inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.]~~ 3

C ~~F~~.1 and C ~~F~~.2 3

If a Required Action of Condition A~~,~~ or B~~, [C], [D] or [E]~~ cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 3

D ~~G~~.1 3

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

lower
particulate

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. ~~[The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~ 2

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

SR 3.4.15.2

lower
particulate

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string. ~~[The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.~~ 2

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

SR 3.4.15.3~~, [SR 3.4.15.4,~~ and SR 3.4.15.4 ~~5]~~ 3

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~[The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3. UFSAR, Section ~~[~~ 5.2.7 ~~]~~. 2 1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. ~~[In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms.]~~ 1

The containment pocket sump used to collect unidentified LEAKAGE ~~[~~is~~]~~ ~~[(or) and the containment air cooler condensate flow rate monitor] [are]~~ instrumented to alarm for increases above the normal flow rates. 2 1

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. A r ~~R~~adioactivity detection system~~s~~ ~~are~~ is included for monitoring ~~both~~ particulate activity ~~and gaseous activities~~ because of ~~their sensitivities~~ its sensitivity and rapid response~~s~~ to RCS LEAKAGE. 2

Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment pocket sump ~~[and condensate flow from air coolers]~~. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO. 2 1

BASES

BACKGROUND (continued)

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. [Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.] 1

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. [Some of these systems could ~~service~~ serve as 6 early alarm systems signaling the ~~operations~~ operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.] 6

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires ~~[three]~~ two instruments to be OPERABLE.

The containment pocket sump is used to collect unidentified LEAKAGE. ~~[The containment sump consists of the normal sump and the emergency sump. The LCO requirements apply to the total amount of unidentified LEAKAGE collected in [the][both] sump[s].]~~ The monitor on the containment pocket sump detects [level ~~or flow rate or the operating frequency of a pump]~~ and is instrumented to detect when there is ~~[leakage of]~~ [an 2 1 2

BASES

LCO (continued)

increase above the normal value by~~]~~ 1 gpm. The identification of ~~[~~an increase in~~]~~ unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment pocket sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment pocket sump level monitor OPERABILITY. 1 2 2

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the lower ~~gaseous or particulate~~ containment atmosphere particulate radioactivity monitor. ~~Only one of the two detectors is required to be OPERABLE.~~ A ~~R~~radioactivity detection system~~s~~ is ~~are~~ included for monitoring ~~both~~ particulate activity ~~and gaseous activities~~ because of the~~ir~~ sensitivity ~~sensitivities~~ and rapid response~~s~~ to RCS LEAKAGE, but ha~~ve~~ as recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the lower ~~gaseous or~~ particulate ~~particulate~~ containment atmosphere radioactivity monitor~~s~~ to detect a 1 gpm increase within 1 hour during normal operation. However, the lower ~~gaseous or~~ particulate ~~particulate~~ containment atmosphere particulate radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitor~~s~~ (Reference 3). 2

~~[An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.]~~ 1

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment pocket sump level monitor, in combination with a ~~gaseous or~~ particulate radioactivity monitor ~~[and a containment air cooler condensate flow rate monitor]~~, provides an acceptable minimum. 2 1

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the required containment pocket sump level monitor inoperable, no other form of sampling can provide the equivalent information; however, the lower containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the lower containment atmosphere particulate radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2 1

Restoration of the required pocket sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1. 2

SII

B.1.1, B.1.2, and B.2 3

~~B.1.1, B.1.2, B.2.1, and B.2.2~~

With ~~both gaseous and particulate~~ the lower containment atmosphere particulate radioactivity monitoring instrumentation channel~~s~~ inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. 2

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required lower containment atmosphere particulate radioactivity monitors. ~~[Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances performed every 24 hours.]~~ 1

BASES

ACTIONS (continued)

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[~~and RCP seal injection and return flows~~]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available. 1

~~[C.1 and C.2~~

~~With the containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.~~ 3

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.]~~

~~D.1, D.2.1, and D.2.2~~

~~With the required containment sump monitor [and the containment air cooler condensate flow rate monitor] inoperable, the only means of detecting LEAKAGE is the required containment atmosphere radiation monitor. A Note clarifies that this Condition is only applicable when the only OPERABLE monitor is the required containment atmosphere gaseous radiation monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the containment atmosphere must be taken to provide alternate periodic information. The 12 hour interval is sufficient to detect increasing RCS~~ 3

BASES

ACTIONS (continued)

~~leakage. The Required Action provides 7 days to restore another RCS leakage monitor to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.~~ 3

~~[E.1 and E.2~~

~~With the required containment atmosphere radioactivity monitor [and the containment air cooler condensate flow rate monitor] inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.]~~ 3

C C
~~F~~.1 and ~~F~~.2 3

or
If a Required Action of Condition A~~,~~ B~~, [C], [D] or [E]~~ cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 3

D
~~G~~.1 3

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

lower particulate

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. ~~[The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~ 2 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 5

SR 3.4.15.2

lower particulate

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string. ~~[The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.~~ 2 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

5

4

SR 3.4.15.3~~, [SR 3.4.15.4,~~ and SR 3.4.15.~~5]~~ 3

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~[The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.~~

~~OR~~

4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

5

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
3. U FSAR, Section ~~[~~ 5.2.7 ~~]~~. 2 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15 BASES, RCS LEAKAGE DETECTION INSTRUMENTATION

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes have been made to be consistent with changes made to the Specifications.

4. ISTS SR 3.4.15.1, ISTS SR 3.4.15.2, ISTS SR 3.4.15.3 and ISTS SR 3.4.15.4 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.4.15.1, ITS SR 3.4.15.2, ITS SR 3.4.15.3 and SR 3.4.15.4 is "In accordance with the Surveillance Frequency Control Program."

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. Editorial/grammatical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 16

ITS 3.4.16, RCS SPECIFIC ACTIVITY

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

ITS A01 ITS 3.4.16

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits L01

LCO 3.4.16 3.4.8 ~~The specific activity of the primary coolant shall be limited to:~~

SR 3.4.16.2 a. Less than or equal to 0.35 microcuries/gram DOSE EQUIVALENT I-131, and

~~b. Less than or equal to 100/Ē microcuries/gram~~.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4 ~~and 5~~

ACTION: L01

MODES 1, 2 and 3~~*~~

Action A a. With ~~the specific activity of the primary coolant greater than 0.35 microcuries/gram~~ DOSE EQUIVALENT I-131 not within limit for more than 48 hours during one continuous time interval or ~~exceeding the limit line shown on Figure 3.4-1~~, be in at least HOT STANDBY ~~with T_{avg} less than 500°F~~ within 6 hours. Add proposed Required Action ~~C~~.2. LCO 3.0.4.c is applicable. L01 L01 M01

Required Action ~~C~~.1

Action A Note

DOSE EQUIVALENT XE-133 not within limits ~~restore within limits in 48 hours.~~ L01

Action B b. With the ~~specific activity of the primary coolant greater than~~ ~~100/Ē~~ ~~microcuries/gram~~, be in at least HOT STANDBY ~~with T_{avg} less than 500°F~~ within 6 hours. ~~Add proposed Action B Note.~~ L01

Required Action ~~C~~.1

Add proposed Required Action ~~C~~.2 M01

RPG-013 B MODES 1, 2, 3, 4 ~~and 5~~ B L01

a. With ~~the specific activity of the primary coolant greater than 0.35 microcuries/gram~~ DOSE EQUIVALENT I-131 not within limit ~~or greater than 100/Ē microcuries/gram~~, perform the sampling and analysis requirements of item 4a of Table 4.4-4 ~~until the specific activity of the primary coolant is restored to within its limits~~. L01 A02

Required Action A.1

~~*With T_{avg} greater than or equal 500°F~~. L01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1
SR 3.4.16.2

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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ITS

A01

ITS 3.4.16

TABLE 4.4-4

In accordance with the Surveillance Frequency Control Program LA01

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Add proposed SR 3.4.16.1 with a Frequency of ~~7 days~~ L01

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. ~~Gross Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3, 4~~
2. ~~Isotopic Analysis for~~ DOSE EQUIVALENT I-131 Concentration	~~1 per 14 days~~ In accordance with the Surveillance Frequency Control Program	1
3. ~~Radiochemical for E Determination~~	~~1 per 6 months*~~	~~1~~
4. ~~Isotopic Analysis for Iodine Including I-131, I-133, and I-135~~	a. Once per 4 hours, whenever ~~the specific activity exceeds 0.35 µCi/gram~~ DOSE EQUIVALENT I-131 not within limit ~~or 100/E µCi/gram,~~ and	1#, 2#, 3#, 4#, ~~5#~~
	b. One sample between 2 and 6 hours following a THERMAL POWER change ~~exceeding~~ ≥ 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3 and 4

L01 LA02 LA01 L01 LA02 A02 L01 L01 L01 A03

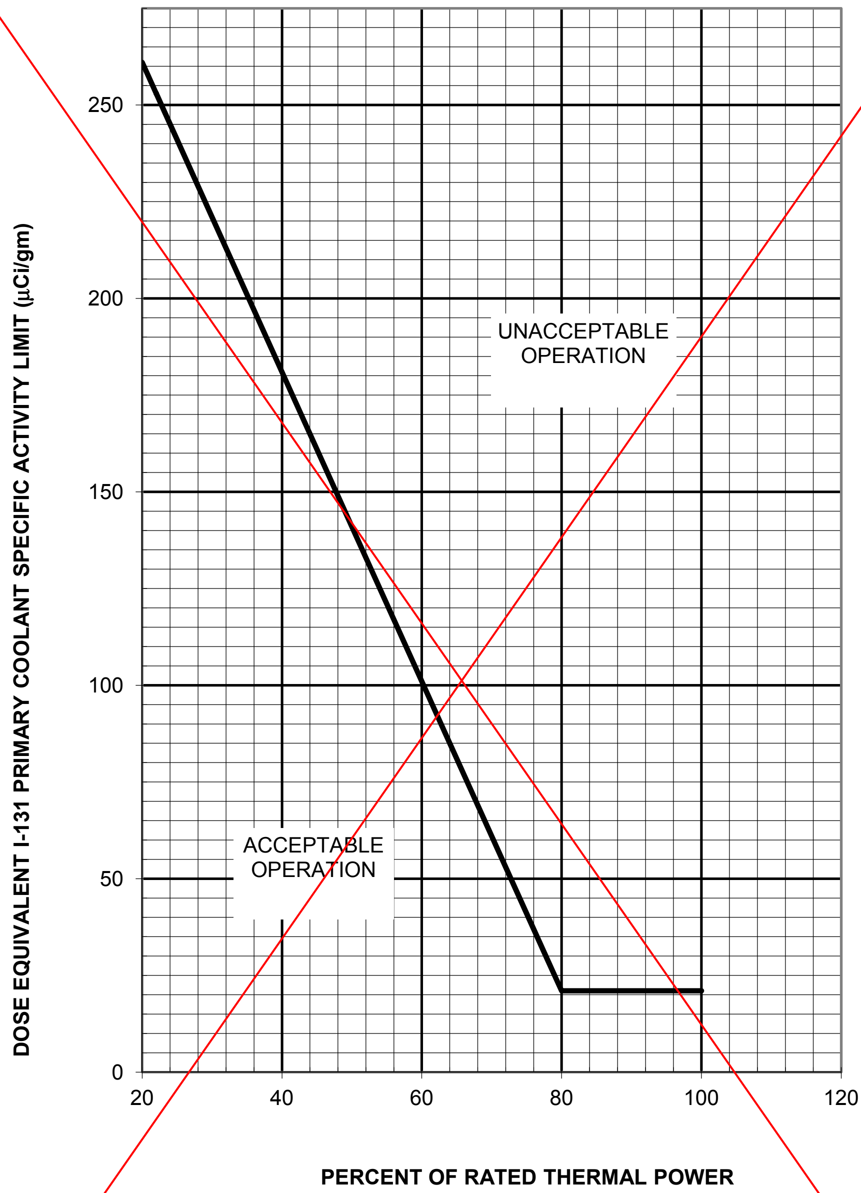
SR 3.4.16.2

Required Action A.1

SR 3.4.16.2

~~#Until the specific activity of the primary coolant system is restored within its limits.~~ A02

~~*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.~~ L01



R241

L01

FIGURE 3.4-1
DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit
Versus
Percent of RATED THERMAL POWER with the Primary Coolant Specific
Activity > 0.35 µCi/gram Dose Equivalent I-131

R241

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REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.16 3.4.8 ~~The specific activity of the primary coolant shall be limited to:~~ RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits L01

SR 3.4.16.2 a. Less than or equal to 0.35 microcurie per gram DOSE EQUIVALENT I-131, and

~~b. Less than or equal to 100/Ē microcuries per gram.~~

Applicability APPLICABILITY: MODES 1, 2, 3, and 4 ~~and 5~~

ACTION:

MODES 1, 2 and 3~~*~~: L01

Action A a. With ~~the specific activity of the primary coolant greater than 0.35 microcurie per gram~~ DOSE EQUIVALENT I-131 not within limit for more than 48 hours during one continuous time interval ~~or exceeding the limit line shown on Figure 3.4-1~~, be in at least HOT STANDBY ~~with T_{avg} less than 500°F~~ within 6 hours. LCO 3.0.4.c is applicable.

Required Action ~~C~~.1

Add proposed Required Action ~~C~~.2. L01

Action A Note

Add proposed Action B

Action B b. With the ~~specific activity of the primary coolant greater than 100/Ē microcurie per gram~~ DOSE EQUIVALENT XE-133 not within limits ~~restore within limits in 48 hours.~~, be in at least HOT STANDBY ~~with T_{avg} less than 500°F~~ within 6 hours. L01

Required Action ~~C~~.1

Add proposed Required Action ~~C~~.2.

B

RPG-013 B

MODES 1, 2, 3, 4 ~~and 5~~: M01

Required Action A.1 a. With ~~the specific activity of the primary coolant greater than 0.35 microcurie per gram~~ DOSE EQUIVALENT I-131 not within limit ~~or greater than 100/Ē microcuries per gram,~~ perform the sampling and analysis requirements of item 4a of Table 4.4-4 ~~until the specific activity of the primary coolant is restored to within its limits~~. L01 A02

~~* With T_{avg} greater than or equal to 500°F~~. L01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1
SR 3.4.16.2
4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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ITS A01 ITS 3.4.16

TABLE 4.4-4

In accordance with the Surveillance Frequency Control Program LA01

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Add proposed SR 3.4.16.1 with a Frequency of ~~7 days~~ L01

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. ~~Gross Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3, 4~~
2. ~~Isotopic Analysis for~~ DOSE EQUIVALENT I-131 Concentration	~~1 per 14 days~~ In accordance with the Surveillance Frequency Control Program	1
~~3. Radiochemical for Ē Determination~~	~~1 per 6 months*~~	~~1~~
4. ~~Isotopic Analysis for Iodine Including I-131, I-133, and I-135~~	a) Once per 4 hours, whenever ~~the specific activity exceeds 0.35 µCi/gram~~ DOSE EQUIVALENT I-131 not within limit ~~or 100/Ē µCi/gram,~~ and	1#, 2#, 3#, 4#, ~~5#~~
	b) One sample between 2 and 6 hours following a THERMAL POWER change ~~exceeding~~ ≥ 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3 and 4

SR 3.4.16.2 (item 2) LA02 LA01 L01 LA02 A02 L01 L01

Required Action A.1 (item 4)

SR 3.4.16.2 (item 4b) L01 A03

~~# Until the specific activity of the primary coolant system is restored within its limits~~ A02

~~* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.~~ L01

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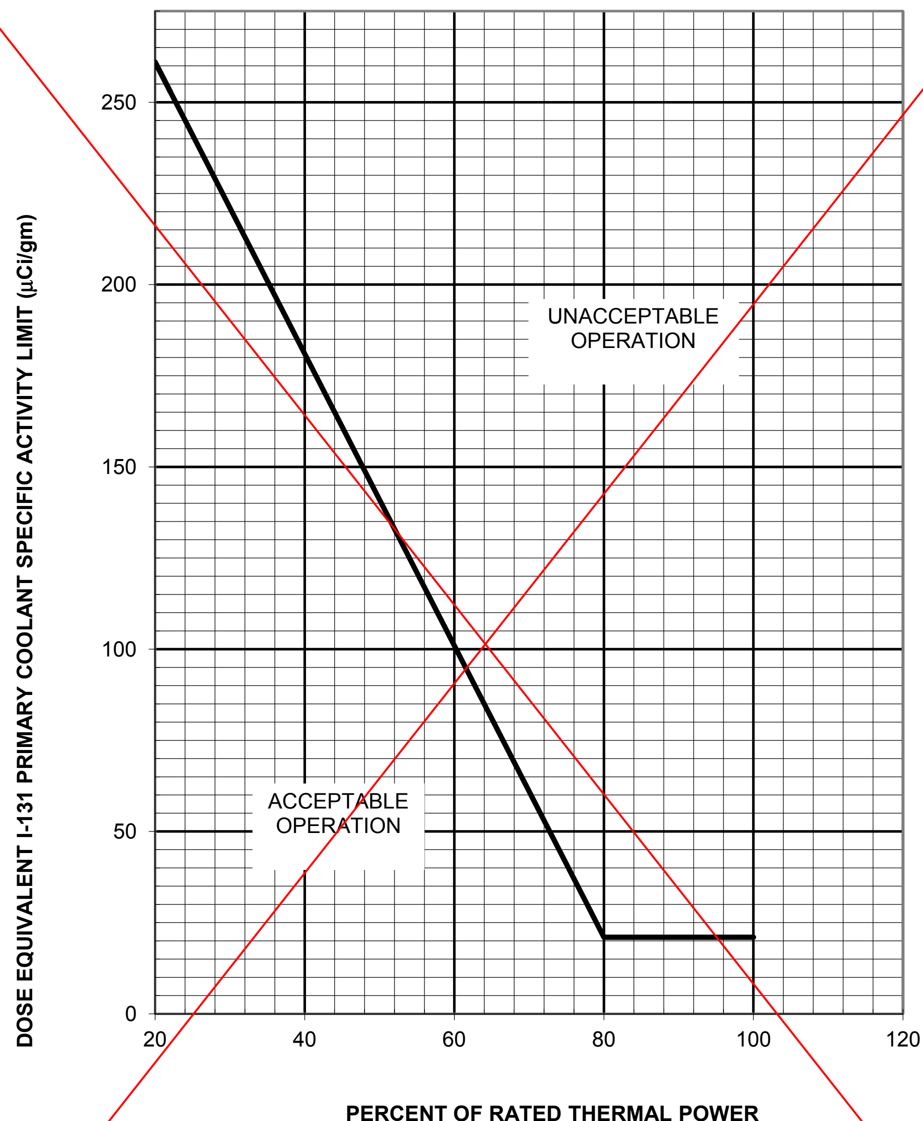
ITS

A01

ITS 3.4.16

L01



R227

FIGURE 3.4-1
DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.35 μCi/gram Dose Equivalent I-131

R227

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DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.4.8 (MODES 1, 2, 3, 4, and 5), ACTION a, requires Table 4.4-4 Sampling Test 4.a, Isotopic Analysis for Iodine, to be performed every 4 hours until the specific activity of the primary coolant system is restored to within limits. This is also restated in CTS Table 4.4-4, Footnote #. ITS 3.4.16 Required Action A.1 essentially requires the same analysis; however the explicit statement to perform the isotopic analysis for iodine "until restored to within limits" is not included. This changes the CTS by deleting the explicit statement to perform the isotopic analysis for iodine until the limits are met.

The purpose of the statement "until the specific activity of the primary coolant is restored to within its limits," in CTS 3.4.8 (MODES 1, 2, 3, 4, and 5) ACTION a and Table 4.4-4 Item 4.a and Footnote #, is to ensure the Surveillance is performed until the limit is met. In ITS, stating that the analysis is required until the specific activity is within limits is unnecessary. ITS LCO 3.0.2 requires the Required Actions of the entered Condition(s) to be performed upon discovery of failure to meet the LCO, until the LCO is met. If the LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. This change is acceptable since ITS LCO 3.0.2 will require the Required Action to be performed until the LCO is met. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS Table 4.4-4 4. b) Frequency requires, that one sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period. ITS SR 3.4.16.2 Frequency requires, one sample between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RTP within 1 hour. This changes the CTS by requiring one sample to be taken between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RTP within 1 hour.

The purpose of CTS Table 4.4-4 4. b) is to ensure one sample is taken between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period. This change is acceptable because ITS SR 3.4.16.2 requires essentially the same Frequency a sample between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RTP within 1 hour. This change is designated as administrative because it does not result in technical changes to CTS.

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DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.8 requires the specific activity of the reactor coolant to be within limit whenever the reactor is in MODES 1, 2, 3, 4 and 5. In addition when a unit shutdown is required in MODES 1, 2 and 3* (Footnote * limits MODE 3 Applicability to Tavg ≥ 500°F) by CTS 3.4.8 ACTION a and CTS 3.4.8 ACTION b, the unit is required to be in HOT STANDBY with T_{avg} less than 500°F within 6 hours. ITS 3.4.16 Applicability, with TSTF-490-A incorporated, requires the RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity to be within limits during MODES 1, 2, 3 and 4. ITS 3.4.16 Required Action ~~C~~.1 requires the unit to be in MODE 3 within 6 hours and Required Action ~~C~~.2 requires the unit to be in MODE 5 within 36 hours. This changes the CTS by relaxing the requirement to be "less than 500°F within 6 hours," to "be in MODE 3 in 6 hours," and by adding Required Action ~~C~~.2 to enter MODE 5 within 36 hours. The change that deletes the E-bar requirement and replaces it with a DOSE EQUIVALENT XE-133 requirement is discussed in DOC L01. B RPG-013

This change is acceptable because the requirement to place the unit in MODE 5 places the unit outside the MODE of Applicability. The Completion Time is based on operating experience and the need to reach the required condition from full power in an orderly manner and without challenging unit systems. This change is designated as more restrictive because it adds a new requirement for the unit to be in MODE 5.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS Table 4.4-4 Item 2 requires, in part, verifying isotopic analysis for DOSE EQUIVALENT I-131 concentration once per 14 days. ITS SR 3.4.16.2 requires a similar Surveillance and specifies the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." Additionally ITS SR 3.4.16.1 has been added to verify reactor coolant DOSE EQUIVALENT XE-133 specific activity within limits at a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." (See DOC L01 for discussion of adding ITS SR 3.4.16.1.) This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the

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DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.4-4 Item 2 requires an isotopic analysis to determine whether DOSE EQUIVALENT I-131 concentration is within limit. CTS Table 4.4-4 Item 4 requires an isotopic analysis for iodine including I-131, I-133, and I-135. ITS SR 3.4.16.2 requires the verification that the reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. ITS 3.4.16 Required Action A.1 requires the verification that DOSE EQUIVALENT I-131 is ≤ 21.0 μCi/gm. This changes the CTS by moving the detail that an Isotopic Analysis or Isotopic Analysis for Iodine including I-131, I-133, and I-135 must be performed to satisfy the requirements of the Surveillance and Action to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because the type of information is not necessary to be included in the Technical Specifications to provide adequate protection to public health and safety. ITS SR 3.4.16.2 and ITS 3.4.16 Required Action A.1 still retain the requirements to verify the reactor coolant DOSE EQUIVALENT I-131 is within limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.8 requires the specific activity of the primary coolant to be less than or equal to 100/Ē μCi/gram. CTS 3.4.8 ACTION b states that if the limit is not met, then the unit must be shut down to HOT STANDBY with T_{avg} less than 500°F within 6 hours – no restoration time prior to the shutdown is provided. Furthermore, if the limit is not met, ACTION a (MODES 1, 2, 3, 4 and 5) requires the sample and analysis requirements of Table TS 4.4-4, item 4.a (an isotopic analysis for iodine), to be performed every 4 hours. Table 4.4-4 Item 3, requires a "Radiochemical for E bar Determination" analysis performed every 6 months with a Footnote limitation (Footnote *) that a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer prior to performance of the analysis. ITS 3.4.16 does not include any requirements related to Ē. ITS LCO 3.4.16 requires the DOSE EQUIVALENT XE-133 limit to be met. SR 3.4.16.1

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DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

states that the DOSE EQUIVALENT XE-133 must be ≤ 1612.6 μCi/gm and only requires ~~of~~ the Surveillance to be performed in MODE S 1, 2, and 3 with $T_{avg} \geq 500^{o}F$. If DOSE EQUIVALENT XE-133 is not within the limit, ITS 3.4.16 ACTION B ~~provides 48 hours to restore the DOSE EQUIVALENT XE-133 to within its limits prior to requiring~~ requires a unit shutdown. ~~It also allows LCO 3.0.4.c to be applicable when in ACTION B~~. Furthermore, when DOSE EQUIVALENT XE-133 is not within its limit, the ITS RPG-013 does not require the isotopic analysis for iodine to be performed every 4 hours. This changes the CTS by deleting the Ē requirements on the primary coolant gross specific activity and replacing it with the DOSE EQUIVALENT XE-133 requirements on primary coolant noble gas activity, consistent with Technical Specification Task Force (TSTF) change traveler TSTF-490-A.

CTS 3.4.8 Applicability for DOSE EQUIVALENT I-131 and E bar is required in MODES 1, 2, 3, 4, and 5. ITS 3.4.16 Applicability for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is required in MODES 1, 2, 3, and 4. This changes the CTS Applicability for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 to MODES 1, 2, 3, and 4, consistent with TSTF-490-A.

The proposed changes are consistent with TSTF-490-A, Revision 0. TSTF-490-A, Revision 0, "Deletion of E Bar definition and Revision to RCS Specific Activity Tech Spec" was announced for availability in the Federal Register on March 15, 2007 as part of the consolidated line item improvement process. The changes were approved by the NRC staff Safety Evaluation (SE) dated March 8, 2007 (ADAMS Accession No. ML070250176). TVA has reviewed the NRC staff SE listed above, the Federal Notice for comment published November 20, 2006 (including the SE), and the Federal Notice for availability published on March 15, 2007. TVA has concluded that the justifications presented in TSTF-490-A, Revision 0 and the model SE prepared by the NRC staff are applicable to SQN and justify this change. The change incorporating the newly defined quantity DOSE EQUIVALENT XE-133 is acceptable from a radiological dose perspective, since it will result in an LCO that more closely relates to non-iodine RCS activity limits to the dose consequence analysis which form the bases. The Dose Conversion Factors used in the determination of DOSE EQUIVALENT I-131 and XE-133 are consistent with Dose Conversion factors used in the applicable dose consequence analysis. This change is less restrictive because the LCO is now being based on noble gas activity ~~and a limited amount of time (48 hours) is provided to restore the limit prior to requiring a unit shutdown~~.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

TSTF-490-A

3.4.8

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits~~ RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

Applicability

APPLICABILITY: MODES ~~1 and 2,~~ 1, 2, 3, and 4
~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F~~.

TSTF-490-A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a (MODES 1, 2, and 3* and MODES 1, 2, 3, 4, and 5) Table 4.4-4 Item 4.a A. DOSE EQUIVALENT I-131 ~~> 1.0 μCi/gm~~ not within limit.	--------------------NOTE------------------- LCO 3.0.4.c is applicable. --- A.1 Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1~~ ≤ ~~[60]~~ 21.0 μCi/gm. AND A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 4 hours 48 hours
~~B. Gross specific activity of the reactor coolant not within limit.~~ DOSE EQUIVALENT XE-133 not within limit	~~B.1 Be in MODE 3 with T_{avg} < 500°F.~~ --------------NOTE----------------- LCO 3.0.4.c is applicable. --- Restore DOSE EQUIVALENT XE-133 to within limit.	~~6~~ 48 hours

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~~TSTF-490-A~~

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ACTIONS (continued)

RPG-013

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C.~~ B. Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR DOSE EQUIVALENT I-131 ~~in the unacceptable region of Figure 3.4.16-1~~ > ~~[60]~~ 21.0 μCi/gm. OR DOSE EQUIVALENT XE-133 not within limit.	~~C.1~~ B.1 Be in MODE 3 ~~with T_{avg} < 500°F~~. ~~AND~~ ~~C.2 Be in MODE 5~~	6 hours ~~36 hours~~

ACTION a (MODES 1, 2, and 3*)

4 TSTF-490-A 1 4

SURVEILLANCE REQUIREMENTS

RPG-013

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	~~Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.~~ -----------------------NOTE----------------------- ~~Only required to be performed in MODE 1.~~ Only required to be performed in MODES 1, 2, and 3 with $T_{avg} \geq$ 500°F --- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ ~~[280]~~ 1612.6 μCi/gm.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.8 Table 4.4-4 Item 1

2 4 TSTF-490-A 2

SURVEILLANCE REQUIREMENTS (continued)

RPG-013

4

	SURVEILLANCE	FREQUENCY
SR 3.4.16.2	~~-----NOTE-----~~ ~~Only required to be performed in MODE 1.~~ ~~----------~~ Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ [~~1.0~~] 0.35 μCi/gm.	~~[14 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program] AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

LCO 3.4.8.a
Table 4.4-4
Item 2

1 2

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Table 4.4-4
Item 4.b

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.16.3~~	~~-----------------------------NOTE----------------------------~~ ~~Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~ ~~---~~ ~~Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	~~[184 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

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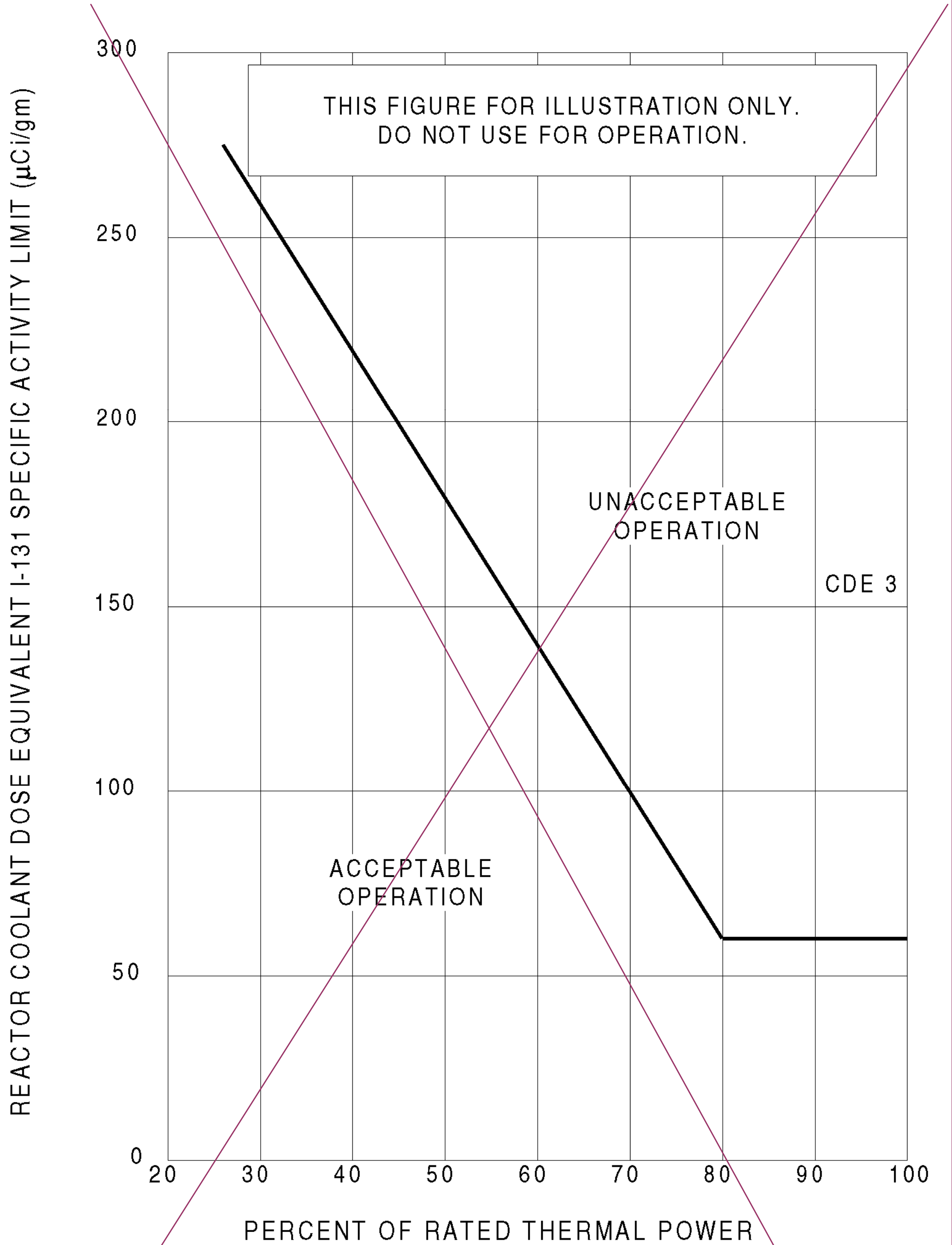

Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

TSTF-490-A

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits

TSTF-490-A

3.4.8 LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits~~.

Applicability APPLICABILITY: MODES ~~1 and 2,~~ 1, 2, 3, and 4
~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F~~.

TSTF-490-A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 ~~> 1.0 μCi/gm~~ not within limit.	--------------------NOTE------------------- LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1~~ ≤ ~~[60]~~ 21.0 μCi/gm. AND	Once per 4 hours
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. ~~Gross specific activity of the reactor coolant not within limit.~~ DOSE EQUIVALENT XE-133 not within limit	--------------NOTE----------------- LCO 3.0.4.c is applicable. -------------------------------------- B.1 ~~Be in MODE 3 with T_{avg} < 500°F.~~ Restore DOSE EQUIVALENT XE-133 to within limit.	~~6~~ 48 hours

ACTION a (MODES 1, 2, and 3* and MODES 1, 2, 3, 4, and 5)

Table 4.4-4 Item 4.a

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TSTF-490-A

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~~TSTF-490-A~~

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CTS

RCS Specific Activity
3.4.16

ACTIONS (continued)

RPG-013

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a (MODES 1, 2, and 3*) B ~~C~~. Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR DOSE EQUIVALENT I-131 ~~in the unacceptable region of Figure 3.4.16-1~~ > ~~[60]~~ 21.0 μCi/gm. OR DOSE EQUIVALENT XE-133 not within limit.	B ~~C~~.1 Be in MODE 3 ~~with T_{avg} < 500°F~~. ~~AND~~ ~~C.2 Be in MODE 5~~	6 hours ~~36 hours~~

4 TSTF-490-A 1

SURVEILLANCE REQUIREMENTS

RPG-013

SURVEILLANCE		FREQUENCY
4.4.8 Table 4.4-4 Item 1 SR 3.4.16.1	~~Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.~~ -----NOTE----- ~~Only required to be performed in MODE 1.~~ Only required to be performed in MODES 1, 2, and 3 with T_{avg} ≥ 500°F ---------- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ ~~[280]~~ 1612.6 μCi/gm.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2 4 TSTF-490-A 2

SURVEILLANCE REQUIREMENTS (continued)

RPG-013

4

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	~~-----NOTE-----~~ ~~Only required to be performed in MODE 1.~~ ~~----------~~	
LCO 3.4.8.a Table 4.4-4 Item 2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ [~~1.0~~ 0.35] μCi/gm.	~~[14 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND
Table 4.4-4 Item 4.b		Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

1 2 TSTF-490-A 2

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.16.3~~	~~------------------------------NOTE------------------------------ Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---~~ ~~Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	~~[184 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

TSTF-490-A

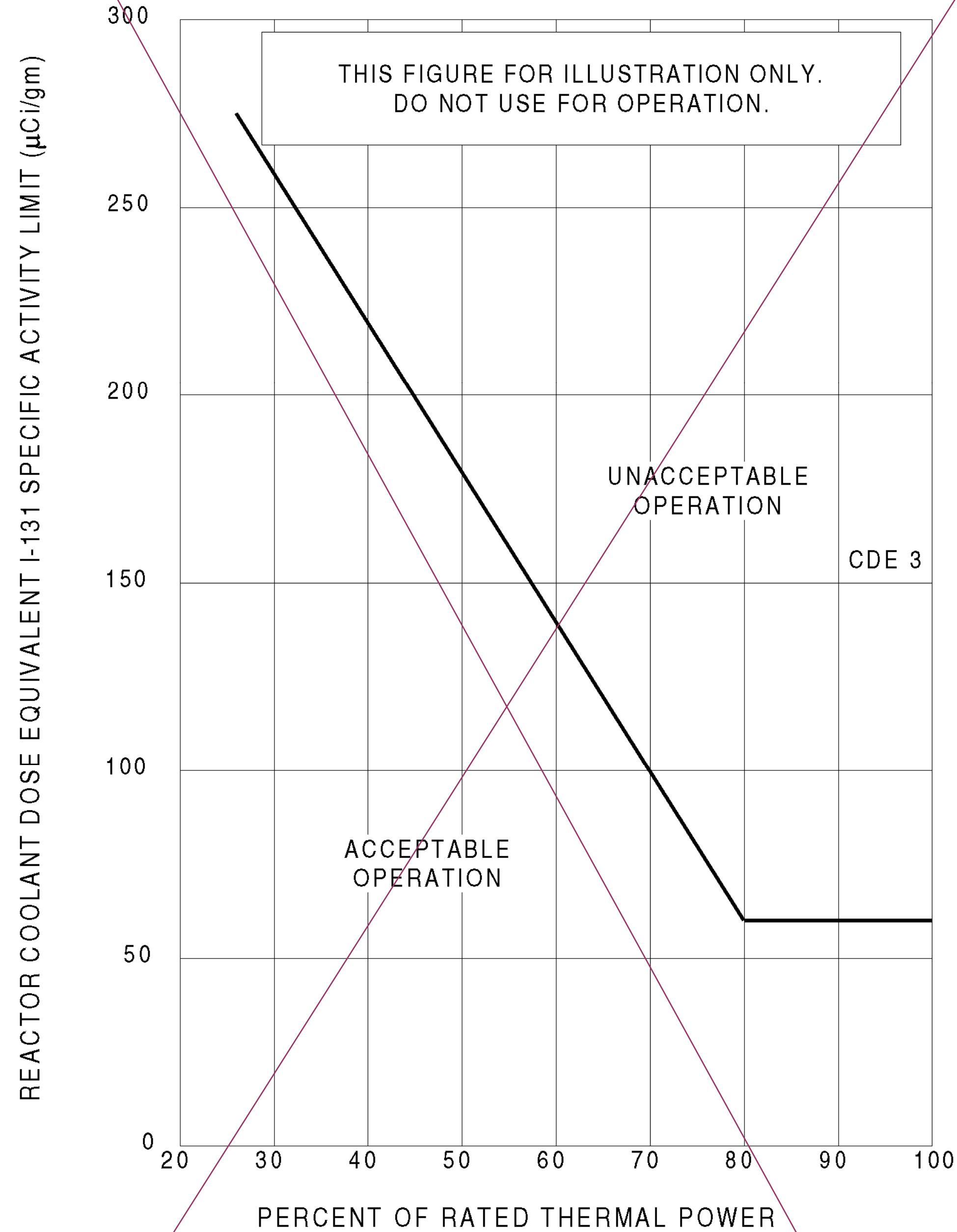

Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

TSTF-490-A

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS SR 3.4.16 (ITS SR 3.4.16.1), and ISTS SR 3.4.16.2 (ITS SR 3.4.16.2) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

RPG-013

4. Changes are made to ISTS 3.4.16 as a result of discussions between NRC staff and TVA during a public meeting held on August 12, 2014. ISTS 3.4.16 Condition B is deleted, ISTS 3.4.16 Condition C is re-sequenced to Condition B and revised to include a third Condition of "OR DOSE EQUIVALENT XE-133 not within limit," and ISTS SR 3.4.16.2 Note associated with DEI is deleted. Additionally, TSTF-490 proposed adding a Note to ISTS SR 3.4.16.1, "Only required to be performed in MODE 1." This Note is revised to state, "Only required to be performed in MODES 1, 2, and 3 with $T_{avg} \geq 500°F$."

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

TSTF-490-A

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

INSERT 1

~~BACKGROUND~~

~~The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.~~

~~The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.~~

~~The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~

~~The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

~~APPLICABLE SAFETY ANALYSES~~

~~The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.6, "Secondary Specific Activity."~~

~~The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.~~

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BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in ~~[~~10 CFR 100.11~~][10 CFR 50.67]~~ (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents. 1

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SRP) (Ref. 2). 5

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of ~~[1 gpm]~~ 150 gpd exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of ~~[~~0.1~~]~~ μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.~~18~~16, "Secondary Specific Activity." 1 1 2

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at ~~[1.0]~~ 0.35 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of ~~335~~ 500), respectively. The second case assumes the initial reactor coolant iodine activity at ~~[60.0]~~ 21.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior 1 2 1

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TSTF-490-A

BASES

APPLICABLE SAFETY ANALYSES (continued)

to the accident. In both cases, the noble gas specific activity is assumed to be ~~[280]~~ 1612.6 μCi/gm DOSE EQUIVALENT XE-133. 1

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal ~~[or an RCS overtemperature ΔT signal]~~. 1

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves ~~[~~and the main steam safety valves~~]~~. 1
The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~[60.0]~~ 21.0 μCi/gm for more than 48 hours. 1

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO	The iodine specific activity in the reactor coolant is limited to ~~[1.0]~~ 0.35 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~[280]~~ 1612.6 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2). 1 1 The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).
APPLICABILITY	In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2). In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

TSTF-490-A

BASES

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.~~

~~The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.~~

~~The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.~~

~~The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.~~

~~The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.~~

~~The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.~~

~~RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~

BASES

~~LCO~~ ~~The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

~~The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.~~

TSTF-490-A

~~APPLICABILITY~~ ~~In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.~~

~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~

ACTIONS A.1 and A.2

An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the ~~limits of Figure 3.4.16-1 are not exceeded~~. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is ~~done to~~ continue to provide a trend.

specific activity is ≤ ~~[60.0]~~ μCi/gm

21.0

d every 4 hours

2 TSTF-490-A 1

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is ~~required, if the limit violation resulted from normal iodine spiking~~.

INSERT 2

TSTF-490-A

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) ~~while~~ relying on ~~the ACTIONS~~.

,

INSERT 3

This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

,

INSERT 2

TSTF-490-A

acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

INSERT 3

TSTF-490-A

Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met.

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BASES

ACTIONS (continued)

B.1 and B.2 6

~~INSERT 4~~

~~With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.~~

~~The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.~~ RPG-013

or if the DOSE EQUIVALENT XE-133 is > 1612.6 µCi/gm,

~~C.1~~ ~~and C.2~~ 6

~~or B~~

If a Required Action and the associated Completion Time of Condition A is not met, or if the DOSE EQUIVALENT I-131 is > ~~[60.0]~~ 21.0 µCi/gm ~~in the unacceptable region of Figure 3.4.16-1~~, the reactor must be brought to MODE 3 ~~with RCS average temperature < 500°F~~ within 6 hours and MODE 5 within 36 hours. The Completion Times ~~of 6 hours is~~ allowed are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems ~~MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems~~. 1 TSTF-490-A

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

INSERT 5

~~SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.~~

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F.~~ ~~[The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.~~ TSTF-490-A 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. TSTF-490-A

INSERT 6

RPG-013

INSERT 4

TSTF-490-A

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

6

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

INSERT 5

TSTF-490-A

SR 3.4.16.1 requires performing a gamma isotopic analysis and calculating the DOSE EQUIVALENT XE-133 using the dose conversion factors in the DOSE EQUIVALENT XE-133 definition ~~once every 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity. 3

INSERT 6

TSTF-490-A

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of ~~DOS~~ DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

RPG-013

A Note modifies the SR to only require the surveillance to be performed in MODES 1, 2, and 3 with $T_{avg} \geq$ 500°F ~~allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~ 6

SURVEILLANCE REQUIREMENTS (continued)

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

4

SR 3.4.16.2

specific activity

This Surveillance is performed ~~in MODE 1 only~~ to ensure iodine remains within limit during normal operation and following fast power changes when ~~fuel failure is more apt to occur~~. ~~[The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days.~~

the LCO

iodine spiking is more apt to occur

TSTF-490-A

3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

4

The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following ~~fuel failure~~; samples at other times would provide inaccurate results.

iodine spiking initiation

TSTF-490-A

~~INSERT 7~~

RPG-013

~~SR 3.4.16.3~~

~~A radiochemical analysis for Ē determination is required with the plant operating in MODE 1 equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. [The Frequency of 184 days recognizes Ē does not change rapidly.~~

TSTF-490-A

RPG-013

INSERT 7 TSTF-490-A

~~The Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

~~This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.~~

TSTF-490-A

REFERENCES

INSERT 8

~~1. 10 CFR 100.11, 1973.~~

~~2. FSAR, Section [15.6.3].~~

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INSERT 8

TSTF-490-A

~~------------------------------ Reviewer's Note --~~
~~The first listed References 1 and 2 are for plants that are licensed to 10 CFR 100.11. The second set of References are for plants that are licensed to 10 CFR 50.67.~~
~~--~~

4

~~[~~ 1. 10 CFR 100.11.

2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).

~~1. 10 CFR 50.67.~~

~~2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."]~~

3. U FSAR, Section ~~[15.1.5]~~ 15.5.4.

4. U FSAR, Section ~~[15.6.3]~~ 15.5.5.

1

2

2

TSTF-490-A

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

INSERT 1

~~BACKGROUND~~

~~The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.~~

~~The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.~~

~~The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~

~~The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

~~APPLICABLE SAFETY ANALYSES~~

~~The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.6, "Secondary Specific Activity."~~

~~The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.~~

INSERT 1 TSTF-490-A

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in ~~[~~10 CFR 100.11~~][10 CFR 50.67]~~ (Ref. 1). Doses to control room 1
operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SRP) (Ref. 2). 5

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam
generator (SG) tube leakage rate of ~~[1 gpm]~~ 150 gpd exists. The safety analyses 1
assume the specific activity of the secondary coolant is at its limit of
~~[~~0.1~~]~~ µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.~~18~~16, "Secondary 1 2
Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific
activity. One case assumes specific activity at ~~[1.0]~~ 0.35 µCi/gm DOSE 1
EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects
to the primary coolant immediately after a SLB (by a factor of 500), or 2
SGTR (by a factor of ~~335~~ 500), respectively. The second case assumes the
initial reactor coolant iodine activity at ~~[60.0]~~ 21.0 µCi/gm DOSE EQUIVALENT 1
I-131 due to an iodine spike caused by a reactor or an RCS transient prior

INSERT 1 TSTF-490-A

BASES

APPLICABLE SAFETY ANALYSES (continued)

to the accident. In both cases, the noble gas specific activity is assumed to be ~~[280]~~ 1612.6 μCi/gm DOSE EQUIVALENT XE-133. 1

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal ~~[or an RCS overtemperature ΔT signal]~~. 1

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves ~~[~~and the main steam safety valves~~]~~. 1 The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~[60.0]~~ 21.0 μCi/gm for more than 48 hours. 1

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

INSERT 1 TSTF-490-A

LCO

The iodine specific activity in the reactor coolant is limited to ~~[1.0]~~ 0.35 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~[280]~~ 1612.6 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2). 1

1

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

TSTF-490-A

BASES

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.~~

~~The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.~~

~~The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.~~

~~The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.~~

~~The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.~~

~~The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.~~

~~RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~

BASES

~~LCO~~

~~The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

~~The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.~~

TSTF-490-A

~~APPLICABILITY~~

~~In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.~~

~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the ~~limits of Figure 3.4.16-1 are not exceeded~~ specific activity is ≤ ~~[60.0]~~ 21.0 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is ~~done to~~ continued every 4 hours to provide a trend.

An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135.

TSTF-490-A 2 1

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is ~~required, if the limit violation resulted from normal iodine spiking~~ INSERT 2.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), ~~while~~ relying on ~~the ACTIONS~~ INSERT 3. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

TSTF-490-A

INSERT 2

TSTF-490-A

acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

INSERT 3

TSTF-490-A

Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met.

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BASES

ACTIONS (continued)

6

B.1 and B.2

~~INSERT 4~~

~~With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.~~

RPG-013

~~The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.~~

or if the DOSE EQUIVALENT XE-133 is > 1612.6 µCi/gm,

6

~~C.1~~ ~~and C.2~~

~~or B~~

21.0

,

> ~~[60.0]~~ µCi/gm

and MODE 5 within 36 hours

are

the required plant conditions from full power conditions in an orderly manner and without challenging plant systems

s

allowed

1

TSTF-490-A

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is ~~in the unacceptable region of Figure 3.4.16-1~~, the reactor must be brought to MODE 3 ~~with RCS average temperature < 500°F~~ within 6 hours. The Completion Time ~~of 6 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

INSERT 5

~~SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.~~

TSTF-490-A

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F.~~ ~~[The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.~~

3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

TSTF-490-A

INSERT 6

2

INSERT 4 TSTF-490-A

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

RPG-013

6

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

INSERT 5 TSTF-490-A

SR 3.4.16.1 requires performing a gamma isotopic analysis and calculating the DOSE EQUIVALENT XE-133 using the dose conversion factors in the DOSE EQUIVALENT XE-133 definition ~~once every 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity. 3

INSERT 6 TSTF-490-A

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of ~~DOS~~ DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

RPG-013

A Note modifies the SR to only require the surveillance to be performed in MODES 1, 2, and 3 with $T_{avg} \geq$ 500°F ~~allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~ 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

4

SR 3.4.16.2

This Surveillance is performed ~~in MODE 1 only~~ to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur ~~fuel failure is more apt to occur~~. ~~[The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days.~~

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3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

4

The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spiking initiation ~~fuel failure~~; samples at other times would provide inaccurate results.

TSTF-490-A

~~INSERT 7~~

RPG-013

~~SR 3.4.16.3~~

~~A radiochemical analysis for Ē determination is required with the plant operating in MODE 1 equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. [The Frequency of 184 days recognizes Ē does not change rapidly.~~

TSTF-490-A

INSERT 7 TSTF-490-A

RPG-013

~~The Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

TSTF-490-A

~~This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.~~

REFERENCES

INSERT 8

~~1. 10 CFR 100.11, 1973.~~

~~2. FSAR, Section [15.6.3].~~

INSERT 8

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~~------------------------------ Reviewer's Note ---------------------------------------~~
~~The first listed References 1 and 2 are for plants that are licensed to 10 CFR 100.11. The second set of References are for plants that are licensed to 10 CFR 50.67.~~
~~---~~

4

~~[~~ 1. 10 CFR 100.11.

2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).

~~1. 10 CFR 50.67.~~

~~2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."]~~

1

3. U FSAR, Section ~~[15.1.5]~~ 15.5.4.

4. U FSAR, Section ~~[15.6.3]~~ 15.5.5.

2

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16 BASES, RCS SPECIFIC ACTIVITY

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.4.16.1 and ISTS SR 3.4.16.2 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.4.16.1 and SR 3.4.16.2 is "In accordance with the Surveillance Frequency Control Program."

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Editorial/grammatical changes to enhance clarity. RPG-013

6. Changes are made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 17

ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

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

ITS — A01 — ITS 3.4.17

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.17 — 3.4.5 SG tube integrity shall be maintained.

AND

plugging — A02

All SG tubes satisfying the tube ~~repair~~ criteria shall be plugged in accordance with the Steam Generator Program.

Applicability — APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS*:

plugging — A02

ACTION A / ACTION B — a. With one or more SG tubes satisfying the tube ~~repair~~ criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

AND

entering MODE 4 — M01

ACTION A — b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to ~~startup~~ following the next refueling outage or SG tube inspection.

add proposed second Condition B entry — M02

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 — 4.4.5.0 Verify steam generator tube integrity in accordance with the Steam Generator Program.

plugging — A02

SR 3.4.17.2 — 4.4.5.1 Verify that each inspected SG tube that satisfies the tube ~~repair~~ criteria is plugged in accordance with the Steam Generator Program prior to ~~startup~~ following a SG tube inspection.

entering MODE 4 — M01

ACTIONS Note — * Separate Action entry is allowed for each SG tube.

Pages 3/4 4-7 through 3/4 4-12 intentionally deleted.

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REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.17 3.4.5 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube ~~repair~~ plugging criteria shall be plugged in accordance with the Steam Generator Program. A02

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

ACTIONS*:

ACTION A a. With one or more SG tubes satisfying the tube ~~repair~~ plugging criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, A02

ACTION B or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

AND

ACTION A b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to ~~startup~~ entering MODE 4 following the next refueling outage or SG tube inspection. M01

add proposed second Condition B entry M02

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 4.4.5.0 Verify steam generator tube integrity in accordance with the Steam Generator Program.

SR 3.4.17.2 4.4.5.1 Verify that each inspected SG tube that satisfies the tube ~~repair~~ plugging criteria is plugged in accordance with the Steam Generator Program prior to ~~startup~~ entering MODE 4 following a SG tube inspection. A02 M01

ACTIONS Note * Separate Action entry is allowed for each SG tube.

Pages 3/4 4-11 through 3/4 4-16 are intentionally deleted

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DISCUSSION OF CHANGES
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 The term “repair” has been replaced with the word “plugging. This is consistent with current approved TSTF-510 and the current SQN licensing basis for repairing steam generator tubes is plugging only.

This change is designated as an administrative change since the change does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.5 ACTION b states to plug the affected tubes(s) in accordance with the Steam Generator Program prior to startup following the next refueling outage or SG tube inspection. CTS 4.4.5.1 states to verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to startup following a SG tube inspection. ITS 3.4.17 ACTION A requires, in part, that SG tubes not meeting the acceptance criteria of the Steam Generator Program be plugged prior to MODE 4 following the next refueling outage or SG tube inspection. This changes the CTS by requiring tube plugging to be completed prior to entering MODE 4 instead of prior to startup.

The purpose of CTS 3.4.5 ACTION b and CTS 4.4.5.1 is to ensure SG tubes not meeting the acceptance criteria of the Steam Generator Program are plugged prior to startup. The ITS 3.4.17, Required Action A.2, Completion Time is required to be performed prior to entering MODE 4 following the next refueling outage or SG tube inspection. This change is designated as more restrictive because it is more prescriptive of the plant condition for which the required plugging needs to be performed.

M02 CTS 3.4.5 does not contain ACTIONS to take should SG tube integrity not be maintained. As a result, LCO 3.0.3 would be entered, which requires action to be initiated within 1 hour, to be in HOT STANDBY (equivalent to ITS MODE 3) within the next 6 hours, in HOT SHUTDOWN within the following, 6 hours, and be in COLD SHUTDOWN (equivalent to ITS MODE 5 within the following 24 hours. Under similar conditions, ITS 3.4.17 ACTION B requires the unit to be in MODE 3 within 6 hours (Required Action B.1) and be in MODE 5 within 36 hours (Required Action B.2). This changes the CTS by providing specific Required Actions to take when SG tube integrity is not maintained instead of requiring

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DISCUSSION OF CHANGES
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

entry into CTS 3.0.3 and thereby reduces the time to reach MODE 5 from 37 hours to 36 hours.

The purpose of requiring a shut down when the SG tube integrity is not maintained is to place the unit in a condition where the differential pressure across the SG tubes is minimized. ITS 3.4.17 ACTION B continues to require the unit to be shut down when SG tube integrity is not maintained. This change is acceptable because the proposed default condition will require the plant to be placed in a condition where the differential pressure across the SG tubes is minimized and the potential for a steam generator tube rupture or steam line break is removed. This change is designated as more restrictive since the 1 hour specified in CTS 3.0.3 no longer applies.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~20~~ 17 Steam Generator (SG) Tube Integrity (1)

3.4.5 LCO 3.4.~~20~~ 17 SG tube integrity shall be maintained. (1)

AND

All SG tubes satisfying the tube plugging ~~[or~~ ~~repair~~] criteria shall be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program. (2) TSTF-510-A

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

ACTIONS

-----NOTE-----

ACTIONS Note * Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ACTION b A. One or more SG tubes satisfying the tube plugging ~~[or~~ ~~repair~~] criteria and not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days (2) (2) TSTF-510-A
	AND	
	A.2 Plug ~~[or repair]~~ the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection (2)
ACTION a B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
OR	B.2 Be in MODE 5.	36 hours
DOC M02 SG tube integrity not maintained.		

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.5.0	SR 3.4.~~20~~.1 17	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program	1
4.4.5.1	SR 3.4.~~20~~.2 17	~~[~~ Verify that each inspected SG tube that satisfies the tube ~~repair~~ plugging ~~[or~~ criteria is plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection	1 2 TSTF-510-A

CTS

SG Tube Integrity
3.4.~~20~~ 17 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~20~~ 17 Steam Generator (SG) Tube Integrity 1

3.4.5 LCO 3.4.~~20~~ 17 SG tube integrity shall be maintained. 1

AND

All SG tubes satisfying the tube plugging ~~[or~~ ~~repair~~] criteria shall be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program. 2 TSTF-510-A

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

ACTIONS

---NOTE---

ACTIONS Note * Separate Condition entry is allowed for each SG tube.

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a ACTION b	A. One or more SG tubes satisfying the tube plugging ~~[or~~ ~~repair~~] criteria and not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days	2 TSTF-510-A 2
		AND		
		A.2 Plug ~~[or repair]~~ the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection	2
ACTION a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours	
		AND		
		B.2 Be in MODE 5.	36 hours	
	OR			
DOC M02	SG tube integrity not maintained.			

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.5.0	SR 3.4.~~20~~ 17 .1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program	1
4.4.5.1	SR 3.4.~~20~~ 17 .2	Verify that each inspected SG tube that satisfies the tube plugging ~~repair~~ criteria is plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection	1 2 TSTF-510-A

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

1. ISTS 3.4.17, "RCS Loop Isolation Valves," ISTS 3.4.18, "RCS Isolated Loop Startup," and ISTS 3.4.19, "RCS Loops – Test Exceptions" have not been adopted. Therefore ISTS 3.4.20, "Steam Generator (SG) Tube Integrity," has been renumbered to ITS 3.4.17, "Steam Generator (SG) Tube Integrity."

2. Sequoyah Unit 1 and 2 are not licensed for repair of SG tubes, so the bracketed allowance has been deleted. ISTS 5.5.9, Steam Generator Program, includes a Reviewer's Note that states the repair criteria currently permitted by plant technical specifications should be provided in the ITS. The bracketed allowance to repair a steam generator tube is not included since the current SQN Steam Generator Program does not allow repair; only plugging is allowed.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~20~~ 17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.~~9~~ 7, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.~~9~~ 7, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.~~9~~ 7. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. (2) (2) (2)

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via ~~safety valves and the majority is discharged to the main condenser~~ SG atmospheric relief valves and safety valves. 3

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of ~~[1 gallon~~ 0.4 gallons per minute~~]~~ or is assumed to increase to ~~[~~1 gallon per minute~~]~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2)~~,~~ and 10 CFR 100 (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~. 3 3

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ plugging ~~[or~~ ~~]~~ criteria be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program. TSTF-510-A 4

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ plugging ~~[or~~ ~~]~~ criteria is ~~[repaired or]~~ removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ plugging ~~[or~~ ~~]~~ criteria but was not plugged ~~[or repaired]~~, the tube may still have tube integrity. TSTF-510-A 4

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall ~~[and any repairs made to it]~~, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. 4

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.~~9~~ 7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. 2

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analys~~i~~es (e) INSERT 1 ~~assumes that accident induced leakage does not exceed [1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. 3 4

3 **INSERT 1**

assumptions are discussed in the Applicable Safety Analyses section.

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BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a ~~main~~ steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative. 3

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

plugging ~~[or~~ ~~]~~ TSTF-510-A

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~ criteria but were not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program as required by SR 3.4.~~20~~.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG ~~repair~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged ~~[or repaired]~~ has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity

4 3 17 plugging ~~[or~~ ~~]~~ 4 TSTF-510-A 4

 1

BASES

ACTIONS (continued)

determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged ~~[or repaired]~~ prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment. 4

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~20~~.1 17 1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging ~~[or~~ ~~repair~~] criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations. TSTF-510-A 3

The Steam Generator Program defines the Frequency of SR 3.4.~~20~~ 17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.~~9~~ 7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.~~9~~ 7 until subsequent inspections support extending the inspection interval. 1 2 TSTF-510-A 2 1

SR 3.4.~~20~~ 17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or~~ ~~repair~~] criteria is ~~[repaired or]~~ removed from service by plugging. The tube plugging ~~[or~~ ~~repair~~] criteria delineated in Specification 5.5.~~9~~ 7 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging ~~[or~~ ~~repair~~] criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. TSTF-510-A 2 4 TSTF-510-A

~~[Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.]~~ 4

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging ~~[or~~ ~~repair~~] criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential. TSTF-510-A 4

1

BASES

REFERENCES	1.	NEI 97-06, "Steam Generator Program Guidelines."
	2.	10 CFR 50 Appendix A, GDC 19.
	3.	10 CFR 100.
	4.	ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
	5.	Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
	6.	EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~20~~ 17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.~~9~~ 7, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.~~9~~ 7, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.~~9~~ 7. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. 2 2 2

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via ~~safety valves and the majority is discharged to the main condenser~~ SG atmospheric relief valves and safety valves. 3

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of ~~[1 gallon~~ 0.4 gallons per minute~~]~~ or is assumed to increase to ~~[~~1 gallon per minute~~]~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2)~~,~~ and 10 CFR 100 (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~. 3

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ plugging ~~[or~~] criteria be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program. TSTF-510-A 4

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ plugging ~~[or~~] criteria is ~~[repaired or]~~ removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ plugging ~~[or~~] criteria but was not plugged ~~[or repaired]~~, the tube may still have tube integrity. TSTF-510-A 4

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall ~~[and any repairs made to it]~~, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. 4

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.~~9~~ 7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. 2

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident e analys~~i~~s INSERT 1 ~~assumes that accident induced leakage does not exceed [1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ 3 4 The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

3 **INSERT 1**

assumptions are discussed in the Applicable Safety Analyses section.

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1

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a ~~main~~ steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative. 3

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

plugging ~~[or~~ ~~]~~ TSTF-510-A

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~ criteria but were not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program as required by SR 3.4.~~20~~.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG ~~repair~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged ~~[or repaired]~~ has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity

4 3 17 plugging ~~[or~~ ~~]~~ 4 TSTF-510-A 4

BASES

ACTIONS (continued)

determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged ~~[or repaired]~~ prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment. 4

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~20~~.1 17 1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging ~~[or~~ ~~repair~~] criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations. TSTF-510-A 3

The Steam Generator Program defines the Frequency of SR 3.4.~~20~~ 17 .1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.~~9~~ 7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.~~9~~ 7 until subsequent inspections support extending the inspection interval. 1 2 TSTF-510-A

SR 3.4.~~20~~ 17 .2 1

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or~~ ~~repair~~] criteria is ~~[repaired or]~~ removed from service by plugging. The tube plugging ~~[or~~ ~~repair~~] criteria delineated in Specification 5.5.~~9~~ 7 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging ~~[or~~ ~~repair~~] criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. TSTF-510-A 2 4 TSTF-510-A

~~[Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.]~~ 4

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging ~~[or~~ ~~repair~~] criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential. TSTF-510-A 4

BASES

REFERENCES	1.	NEI 97-06, "Steam Generator Program Guidelines."
	2.	10 CFR 50 Appendix A, GDC 19.
	3.	10 CFR 100.
	4.	ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
	5.	Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
	6.	EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17 BASES, STEAM GENERATOR (SG) TUBE INTEGRITY

1. ISTS 3.4.17, "RCS Loop Isolation Valves," ISTS 3.4.18, "RCS Isolated Loop Startup," and ISTS 3.4.19, "RCS Loops – Test Exceptions" have not been adopted. Therefore, ISTS 3.4.20, "Steam Generator (SG) Tube Integrity," has been renumbered to ITS 3.4.17, "Steam Generator (SG) Tube Integrity."

2. Changes have been made to reflect changes made to other Specifications.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 18

Improved Standard Technical Specifications (ISTS) Not Adopted in the Sequoyah ITS

ISTS 3.4.17, RCS LOOP ISOLATION VALVES

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

~~3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~3.4.17 RCS Loop Isolation Valves~~

~~LCO 3.4.17~~ ~~Each RCS hot and cold leg loop isolation valve shall be open with power removed from each isolation valve operator.~~

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

~~ACTIONS~~

~~---NOTE---~~
~~Separate Condition entry is allowed for each RCS loop isolation valve.~~
~~---~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. Power available to one or more loop isolation valve operators.~~	~~A.1 Remove power from loop isolation valve operators.~~	~~30 minutes~~
~~B. ------------NOTE------------ All Required Actions shall be completed whenever this Condition is entered. --------------------------------~~ ~~One or more RCS loop isolation valves closed.~~	~~B.1 Maintain valve(s) closed.~~ ~~AND~~ ~~B.2 Be in MODE 3.~~ ~~AND~~ ~~B.3 Be in MODE 5.~~	~~Immediately~~ ~~6 hours~~ ~~36 hours~~

1

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.4.17.1~~	~~Verify each RCS loop isolation valve is open and power is removed from each loop isolation valve operator.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

1

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, RCS LOOP ISOLATION VALVES

1. ISTS 3.4.17, "RCS Loop Isolation Valves," is not included in the Sequoyah Nuclear Plant (SQN) ITS because the RCS system hot and cold loops do not include isolation valves.

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

~~B 3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~B 3.4.17 RCS Loop Isolation Valves~~

~~BASES~~

~~BACKGROUND~~

~~The reactor coolant loops are equipped with loop isolation valves that permit any loop to be isolated from the reactor vessel. One valve is installed on each hot leg and one on each cold leg. The loop isolation valves are used to perform maintenance on an isolated loop. Power operation with a loop isolated is not permitted.~~

~~To ensure that inadvertent closure of a loop isolation valve does not occur, the valves must be open with power to the valve operators removed in MODES 1, 2, 3, and 4. If the valves are closed, a set of administrative controls and equipment interlocks must be satisfied prior to opening the isolation valves as described in LCO 3.4.18, "RCS Isolated Loop Startup."~~

~~APPLICABLE SAFETY ANALYSES~~

~~The safety analyses performed for the reactor at power assume that all reactor coolant loops are initially in operation and the loop isolation valves are open. This LCO places controls on the loop isolation valves to ensure that the valves are not inadvertently closed in MODES 1, 2, 3, and 4. The inadvertent closure of a loop isolation valve when the Reactor Coolant Pumps (RCPs) are operating will result in a partial loss of forced reactor coolant flow (Ref. 1). If the reactor is at power at the time of the event, the effect of the partial loss of forced coolant flow is a rapid increase in the coolant temperature which could result in DNB with subsequent fuel damage if the reactor is not tripped by the Low Flow reactor trip. If the reactor is shutdown and an RCS loop is in operation removing decay heat, closure of the loop isolation valve associated with the operating loop could also result in increasing coolant temperature and the possibility of fuel damage.~~

~~RCS Loop Isolation Valves satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~

~~LCO~~

~~This LCO ensures that the loop isolation valves are open and power to the valve operators is removed. Loop isolation valves are used for performing maintenance in MODES 5 and 6. The safety analyses assume that the loop isolation valves are open in any RCS loops required to be OPERABLE by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," or LCO 3.4.6, "RCS Loops – MODE 4."~~

1

~~BASES~~

~~APPLICABILITY~~

~~In MODES 1 through 4, this LCO ensures that the loop isolation valves are open and power to the valve operators is removed. The safety analyses assume that the loop isolation valves are open in any RCS loops required to be OPERABLE.~~

~~In MODES 5 and 6, the loop isolation valves may be closed. Controlled startup of an isolated loop is governed by the requirements of LCO 3.4.18, "RCS Isolated Loop Startup."~~

~~ACTIONS~~

~~The Actions have been provided with a Note to clarify that all RCS loop isolation valves for this LCO are treated as separate entities, each with separate Completion Times, i.e., the Completion Time is on a component basis.~~

~~A.1~~

~~If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The Completion Time of 30 minutes to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.~~

~~B.1, B.2, and B.3~~

~~Should a loop isolation valve be closed in MODES 1 through 4, the affected loop must be fully isolated immediately and the plant placed in MODE 5. Once in MODE 5, the isolated loop may be started in a controlled manner in accordance with LCO 3.4.18, "RCS Isolated Loop Startup." Opening the closed isolation valve in MODES 1 through 4 could result in colder water or water at a lower boron concentration being mixed with the operating RCS loops resulting in positive reactivity insertion. The Completion Time of Required Action B.1 allows time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

1

~~BASES~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.4.17.1~~

~~The Surveillance is performed to ensure that the RCS loop isolation valves are open, with power removed from the loop isolation valve operators. The primary function of this Surveillance is to ensure that power is removed from the valve operators, since SR 3.4.4.1 of LCO 3.4.4, "RCS Loops - MODES 1 and 2," ensures that the loop isolation valves are open by verifying every 12 hours that all loops are operating and circulating reactor coolant. [The Frequency of 31 days ensures that the required flow can be made available, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day Frequency is justified.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

~~REFERENCES~~ ~~1. FSAR, Section [15.2.6].~~

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17 BASES, RCS LOOP ISOLATION VALVES

1. ISTS 3.4.17 Bases, "RCS Loop Isolation Valves," is not included in the Sequoyah Nuclear Plant (SQN) ITS because the RCS system hot and cold loops do not include isolation valves.

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ISTS 3.4.18, RCS ISOLATION LOOP STARTUP

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

~~3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~3.4.18 RCS Isolated Loop Startup~~

~~LCO 3.4.18 Each RCS isolated loop shall remain isolated with:~~

~~a. The hot and cold leg isolation valves closed if boron concentration of the isolated loop is less than boron concentration required to meet the SDM of LCO 3.1.1 or boron concentration of LCO 3.9.1 and~~

~~b. The cold leg isolation valve closed if the cold leg temperature of the isolated loop is > [20]°F below the highest cold leg temperature of the operating loops.~~

~~APPLICABILITY: MODES 5 and 6.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. Isolated loop hot or cold leg isolation valve open with LCO requirements not met.~~	~~A.1 -------------NOTE------------- Only required if boron concentration requirement not met. ---------------------------------~~	
	~~Close hot and cold leg isolation valves.~~	~~Immediately~~
	~~OR~~	
	~~A.2 -------------NOTE------------- Only required if temperature requirement not met. ---------------------------------~~	
	~~Close cold leg isolation valve.~~	~~Immediately~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.4.18.1~~	~~Verify cold leg temperature of isolated loop is ≤ [20]°F below the highest cold leg temperature of the operating loops.~~	~~Within 30 minutes prior to opening the cold leg isolation valve in isolated loop~~
~~SR 3.4.18.2~~	~~Verify boron concentration of isolated loop is greater than or equal to the boron concentration required to meet the SDM of LCO 3.1.1 or boron concentration of LCO 3.9.1.~~	~~Within 2 hours prior to opening the hot or cold leg isolation valve in isolated loop~~

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.4.18, RCS ISOLATION LOOP STARTUP

1. ISTS 3.4.18, RCS Isolation Loop Startup is not being adopted in the Sequoyah Nuclear Plant (SQN) ITS because the RCS hot and cold leg loops do not include isolation valves. Therefore, ISTS 3.4.18 is not included in the ITS.

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

~~B 3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~B 3.4.18 RCS Isolated Loop Startup~~

~~BASES~~

~~BACKGROUND~~

~~The RCS may be operated with loops isolated in MODES 5 and 6 in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SDM if:~~

~~a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident) or~~

~~b. The boron concentration in the isolated loop is lower than the boron concentration required to meet the SDM of LCO 3.1.1 or boron concentration of LCO 3.9.1 (boron dilution incident).~~

~~As discussed in the FSAR (Ref. 1), the startup of an isolated loop is done in a controlled manner that virtually eliminates any sudden reactivity addition from cold water or boron dilution because:~~ 1

~~a. This LCO and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops,~~

~~b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot leg and cold leg of the isolated loop are within 20°F of the operating loops. Compliance with the temperature requirement is ensured by operating procedures and automatic interlocks, and~~

~~c. Other automatic interlocks prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed. All of the interlocks are part of the Reactor Protection System.~~

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~~BASES~~

~~APPLICABLE SAFETY ANALYSES~~	~~During startup of an isolated loop, the cold leg loop isolation valve interlocks and operating procedures prevent opening the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur.~~ ~~The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents. Violation of this LCO could result in the SDM being reduced in the operating loops to less than that assumed in the safety analyses.~~ ~~The boron concentration of an isolated loop may affect SDM and therefore RCS isolated loop startup satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~
~~LCO~~	~~Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6. This LCO ensures that the loop isolation valves remain closed until the differentials of temperature and boron concentration between the operating loops and the isolated loops are within acceptable limits.~~
~~APPLICABILITY~~	~~In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. Controlled startup of isolated loops is possible without significant risk of inadvertent criticality. This LCO is applicable under these conditions.~~
~~ACTIONS~~	~~A.1 and A.2~~ ~~Required Action A.1 and Required Action A.2 assume that the prerequisites of the LCO are not met and a loop isolation valve has been inadvertently opened. Therefore, the Actions require immediate closure of isolation valves to preclude a boron dilution event or a cold water event. However, each Required Action is preceded by a Note that states that Action is required only when a specific concentration or temperature requirement is not met.~~
~~SURVEILLANCE REQUIREMENTS~~	~~SR 3.4.18.1~~ ~~This Surveillance is performed to ensure that the temperature differential between the isolated loop and the operating loops is ≤ [20]°F. Performing the Surveillance 30 minutes prior to opening the cold leg isolation valve in the isolated loop provides reasonable assurance, based~~

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~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~on engineering judgment, that the temperature differential will stay within limits until the cold leg isolation valve is opened. This Frequency has been shown to be acceptable through operating experience.~~

~~SR 3.4.18.2~~

~~To ensure that the boron concentration of the isolated loop is greater than or equal to the boron concentration required to meet the SDM of LCO 3.1.1 or boron concentration of LCO 3.9.1, a Surveillance is performed 2 hours prior to opening either the hot or cold leg isolation valve. Performing the Surveillance 2 hours prior to opening either the hot or cold leg isolation valve provides reasonable assurance the boron concentration difference will stay within acceptable limits until the loop is unisolated. This Frequency has been shown to be acceptable through operating experience.~~

~~REFERENCES~~ ~~1. FSAR, Section [15.2.6].~~

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.18 BASES, RCS ISOLATION LOOP STARTUP

1. ISTS 3.4.18 Bases, RCS Isolation Loop Startup is not being adopted in the Sequoyah Nuclear Plant (SQN) ITS because the RCS hot and cold leg loops do not include isolation valves. Therefore, ISTS 3.4.18 is not included in the ITS.

ISTS 3.4.19, RCS LOOPS – TEST EXCEPTIONS

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

~~3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~3.4.19 RCS Loops - Test Exceptions~~

~~LCO 3.4.19 The requirements of LCO 3.4.4, "RCS Loops - MODES 1 and 2," may be suspended with THERMAL POWER < P-7.~~

~~APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. THERMAL POWER ≥ P-7.~~	~~A.1 Open reactor trip breakers.~~	~~Immediately~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~SR 3.4.19.1 Verify THERMAL POWER is < P-7.~~	~~[1 hour~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
~~SR 3.4.19.2 Perform a COT for each power range neutron flux - low channel, intermediate range neutron flux channel, P-10, and P-13.~~	~~Prior to initiation of startup and PHYSICS TESTS~~
~~SR 3.4.19.3 Perform an ACTUATION LOGIC TEST on P-7.~~	~~Prior to initiation of startup and PHYSICS TESTS~~

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JUSTIFICATION FOR DEVIATIONS
ISTS 3.4.19, RCS LOOPS – TEST EXCEPTIONS

1. ISTS 3.4.19, "RCS Loops – Test Exceptions," is not included in the Sequoyah Nuclear Plant (SQN) ITS because the exception is not needed to perform any required startup or PHYSICS TESTS.

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

~~B 3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~B 3.4.19 RCS Loops - Test Exceptions~~

~~BASES~~

~~BACKGROUND~~

~~The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, "Quality Standards and Records" (Ref. 2).~~

~~The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.~~

~~The tests will include verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 20% RTP, performing natural circulation cooldown on emergency power, and during the cooldown, showing that adequate boron mixture occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.~~

~~APPLICABLE SAFETY ANALYSES~~

~~The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability.~~

~~As describe in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.~~

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~~BASES~~

~~LCO~~

~~This LCO provides an exemption to the requirements of LCO 3.4.4.~~

~~The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed.~~

~~In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 25% RTP. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits.~~

~~The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.~~

~~APPLICABILITY~~

~~This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.~~

~~ACTIONS~~

~~A.1~~

~~When THERMAL POWER is ≥ the P-7 interlock setpoint 10%, the only acceptable action is to ensure the reactor trip breakers (RTBs) are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the RTBs will shut down the reactor and prevent operation of the fuel outside of its design limits.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.4.19.1~~

~~Verification that the power level is < the P-7 interlock setpoint (10%) will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. [The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.~~

1

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

~~SR 3.4.19.2~~

~~The power range and intermediate range neutron detectors. P-10, and the P-13 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. The Low Power Reactor Trips Block, P-7 interlock, is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Chamber Pressure, P-13 interlock. The P-7 interlock is a logic Function with train, not channel identity. A COT is performed prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The SR 3.3.1.8 Frequency is sufficient for the power range and intermediate range neutron detectors to ensure that the instrumentation is OPERABLE before initiating PHYSICS TESTS.~~

~~SR 3.4.19.3~~

~~The Low Power Reactor Trips Block, P-7 interlock, must be verified to be OPERABLE in MODE 1 by LCO 3.3.1, "Reactor Trip System Instrumentation." The P-7 interlock is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Chamber Pressure, P-13 interlock. The P-7 interlock is a logic Function. An ACTUATION~~

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~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~LOGIC TEST is performed to verify OPERABILITY of the P-7 interlock prior to initiation of startup and PHYSICS TESTS. This will ensure that the RTS is properly functioning to provide the required degree of core protection during the performance of the PHYSICS TESTS.~~

~~REFERENCES~~

~~1. 10 CFR 50, Appendix B, Section XI.~~

~~2. 10 CFR 50, Appendix A, GDC 1, 1988.~~

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JUSTIFICATION FOR DEVIATIONS
ITS 3.4.19 BASES, RCS LOOPS – TEST EXCEPTIONS

1. ISTS 3.4.19 Bases, "RCS Loops – Test Exceptions," is not included in the Sequoyah Nuclear Plant (SQN) ITS because the exception is not needed to perform any required startup or PHYSICS TESTS.

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ENCLOSURE 2

VOLUME 10

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.5
EMERGENCY CORE COOLING SYSTEMS (ECCS)

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.5.1, ACCUMULATORS

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

ITS | A01 | ITS 3.5.1

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 — 3.5.1.1 Four ECCS ~~Each~~ cold leg injection accumulator shall be OPERABLE with: A02

SR 3.5.1.1 — a. The isolation valve open,

SR 3.5.1.2 — b. A contained borated water volume of between 7615 and 7960 gallons of borated water,

SR 3.5.1.4 — c. Between 2400 and 2700 ppm of boron,

SR 3.5.1.3 — d. A nitrogen cover-pressure of between 624 and 668 psig, and

SR 3.5.1.5 — e. Power removed from isolation valve when RCS pressure is ~~above~~ greater than or equal to 2000 psig. A05

Applicability — APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

ACTION B / ACTION C — a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and reduce ~~pressurizer~~ RCS pressure to 1000 psig or less within the following 6 hours. A04

ACTION A / ACTION C — b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce ~~pressurizer~~ RCS pressure to 1000 psig or less within the following 6 hours. A04

Add proposed ACTION D A03

Applicability — *~~Pressurizer~~ RCS pressure above 1000 psig. A04

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ITS

A01

ITS 3.5.1

EMERGENCY CORE COOLING SYSTEMS (ECCS)

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program LA01

SR 3.5.1.1
SR 3.5.1.2
SR 3.5.1.3

a. ~~At least once per 12 hours~~ by:

SR 3.5.1.2
SR 3.5.1.3

1. Verifying the contained borated water volume and nitrogen cover-pressure in each cold leg injection accumulator, and

SR 3.5.1.1

2. Verifying that each cold leg injection accumulator isolation valve is fully open.

In accordance with the Surveillance Frequency Control Program LA01

SR 3.5.1.4

b. ~~At least once per 31 days~~ and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume, that is not the result of addition from the refueling water storage tank, # by verifying the boron concentration of the cold leg injection accumulator solution.

In accordance with the Surveillance Frequency Control Program LA01

SR 3.5.1.5

c. ~~At least once per 31 days~~ when the RCS pressure is ~~above~~ greater than or equal to 2000 psig by verifying that power to the isolation valve operator is removed. A05

SR 3.5.1.4 # Only required to be performed for affected accumulators that experienced volume increases.

ITS A01 ITS 3.5.1

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1.1 Four ECCS ~~Each~~ cold leg injection accumulator shall be OPERABLE with: A02

SR 3.5.1.1 a. The isolation valve open,

SR 3.5.1.2 b. A contained borated water volume of between 7615 and 7960 gallons of borated water,

SR 3.5.1.4 c. Between 2400 and 2700 ppm of boron,

SR 3.5.1.3 d. A nitrogen cover-pressure of between 624 and 668 psig, and

SR 3.5.1.5 e. Power removed from isolation valve when RCS pressure is greater than or equal ~~above~~ 2000 psig. A05

Applicability APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

ACTION B / ACTION C a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and reduce RCS ~~pressurizer~~ pressure to 1000 psig or less within the following 6 hours. A04

ACTION A / ACTION C b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce RCS ~~pressurizer~~ pressure to 1000 psig or less within the following 6 hours. A04

Add proposed ACTION D A03

Applicability * RCS ~~Pressurizer~~ pressure above 1000 psig. A04

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

SR 3.5.1.1 SR 3.5.1.2 SR 3.5.1.3 a. ~~At least once per 12 hours~~ In accordance with the Surveillance Frequency Control Program by: LA01

SR 3.5.1.2 SR 3.5.1.3 1. Verifying the contained borated water volume and nitrogen cover-pressure each cold leg injection accumulator, and

SR 3.5.1.1 2. Verifying that each cold leg injection accumulator isolation valve is fully open.

SR 3.5.1.4 b. ~~At least once per 31 days~~ In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume, that is not the result of addition from the refueling water storage tank,# by verifying the boron concentration of the cold leg injection accumulator solution. LA01

SR 3.5.1.5 c. ~~At least once per 31 days~~ In accordance with the Surveillance Frequency Control Program when the RCS pressure is ~~above~~ greater than or equal 2000 psig by verifying that power to the isolation valve operator is removed. LA01 A05

SR 3.5.1.4 # Only required to be performed for affected accumulators that experience volume increases.

DISCUSSION OF CHANGES
ITS 3.5.1, ACCUMULATORS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.5.1.1 requires "each" cold leg injection accumulator to be OPERABLE. ITS LCO 3.5.1 requires "four" ECCS accumulators to be OPERABLE. This changes the CTS by specifying the exact number of ECCS accumulators required to be OPERABLE.

The change is acceptable because the total number of ECCS accumulators in each unit at SQN is four. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.5.1.1 does not contain a specific ACTION for two or more accumulators inoperable. With two or more accumulators inoperable, CTS 3.0.3 would be entered. ITS 3.5.1 ACTION D directs entry into LCO 3.0.3 when two or more accumulators are inoperable. This changes the CTS by specifically stating to enter LCO 3.0.3 in this System Specification.

This change is acceptable because the action taken when two or more accumulators are inoperable is unchanged. Adding this ACTION is consistent with the ITS convention of directing entry into LCO 3.0.3 when multiple ACTIONS are presented in the ITS, and entry into these multiple ACTIONS could result in a loss of safety function. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.5.1.1 ACTION a requires, with one cold leg accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours. CTS ACTION b requires, with one cold leg accumulator inoperable due to boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours. ITS 3.5.1 ACTION C Required Action C.2 requires, to reduce the RCS pressure to ≤ 1000 psig. This changes the CTS by replacing "pressurizer pressure" with "RCS pressure".

The purpose of CTS 3.5.1.1 ACTIONS a and b provide compensatory measures for an inoperable accumulator. The change to reduce RCS pressure to ≤ 1000 psig better reflects the deviation to the Specification. This change is acceptable because the difference between the pressurizer pressure and RCS pressure is

DISCUSSION OF CHANGES
ITS 3.5.1, ACCUMULATORS

not significant. This change is designated as administrative because it does not result in a technical change to the CTS.

A05 CTS 3.5.1.1.e requires that each cold leg injection accumulator be OPERABLE with power removed from the isolation valve when RCS pressure is above 2000 psig. CTS 4.5.1.1.1.c requires verification that power is removed from the isolation valve operator once per 31 days when RCS pressure is above 2000 psig. ITS SR 3.5.1.5 requires verifying power is removed from each accumulator isolation valve operator when RCS pressure is greater than or equal to 2000 psig. This changes the CTS by requiring power removed from the accumulator isolation valve operators when RCS pressure is at or above 2000 psig.

The purpose of CTS 3.5.1.1.e and CTS 4.5.1.1.1.c is to ensure power is removed from each accumulator isolation valve when RCS pressure is above 2000 psig. This change is acceptable because ITS SR 3.5.1.5 requires essentially the same verification of power to be removed from each accumulator isolation valve operator when RCS pressure is greater than or equal to 2000 psig. This change is designated as administrative because it does not result in technical changes to CTS.

<u>MORE RESTRICTIVE CHANGES</u>

None

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.5.1.1.1.a.1 requires verification of the contained borated water volume and nitrogen cover pressure in each cold leg injection accumulator at least once per 12 hours. CTS 4.5.1.1.1.a.2 requires verification of each cold leg injection accumulator isolation valve is fully open at least once per 12 hours. CTS 4.5.1.1.1.b requires verification of the boron concentration of the cold leg accumulator solution at least once per 31 days. CTS 4.5.1.1.1.c requires verification that power to the isolation valve operator is removed when the RCS pressure is above 2000 psig at least once per 31 days. ITS SR 3.5.1.1, SR 3.5.1.2, SR 3.5.1.3, SR 3.5.1.4, and SR 3.5.1.5 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information

DISCUSSION OF CHANGES
ITS 3.5.1, ACCUMULATORS

is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

3.5.1.1 LCO 3.5.1 ~~[~~Four~~]~~ ECCS accumulators shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > ~~[~~1000~~]~~ psig. 1

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b	A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
ACTION a	B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
ACTION a ACTION b	C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ ~~[~~1000~~]~~ psig. (1)	6 hours 12 hours
DOC A03	D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
3.5.1.1.a 4.5.1.1.1.a.2	SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
3.5.1.1.b 4.5.1.1.1.a.1	SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ ~~[~~ ~~7853~~ 7615 gallons ~~( )%~~ and ≤ ~~8171~~ 7960 gallons ~~( )%].~~	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2
3.5.1.1.d 4.5.1.1.1.a.1	SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ ~~[~~~~385~~ 624 ~~]~~ psig and ≤ ~~[~~~~481~~ 668 ~~]~~ psig.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
3.5.1.1.c 4.5.1.1.1.b	SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ ~~[1900]~~ 2400 ppm and ≤ ~~[2100]~~ 2700 ppm.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND --------NOTE-------- Only required to be performed for affected accumulators ----------------------- Once within 6 hours after each solution volume increase of ≥ 1% of tank volume ~~[[ ] gallons, ( )% of indicated level]~~ that is not the result of addition from the refueling water storage tank	2 1 2 3 1
3.5.1.1.e 4.5.1.1.1.c	SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ ~~[~~2000~~]~~ psig.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

3.5.1.1 LCO 3.5.1 ~~[~~Four~~]~~ ECCS accumulators shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > ~~[~~1000~~]~~ psig. 1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION b	A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours	
ACTION a	B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours	
ACTION a ACTION b	C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ ~~[~~1000~~]~~ psig.	6 hours 12 hours	1
DOC A03	D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately	

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CTS

Accumulators
3.5.1

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY
3.5.1.1.a 4.5.1.1.1.a.2	SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
3.5.1.1.b 4.5.1.1.1.a.1	SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ ~~[7853~~ 7615 gallons ~~( )%~~ and ≤ ~~8171~~ 7960 gallons ~~( )%].~~	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2 1
3.5.1.1.d 4.5.1.1.1.a.1	SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ ~~[385]~~ 624 psig and ≤ ~~[481]~~ 668 psig.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2 1

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
3.5.1.1.c 4.5.1.1.1.b	SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ ~~[1900]~~ 2400 ppm and ≤ ~~[2100]~~ 2700 ppm.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND --------NOTE-------- Only required to be performed for affected accumulators ------------------------ Once within 6 hours after each solution volume increase of ≥ 1% of tank volume ~~[[ ] gallons, ( )% of indicated level]~~ that is not the result of addition from the refueling water storage tank	2 1 2 3 1
3.5.1.1.e 4.5.1.1.1.c	SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ ~~[~~2000~~]~~ psig.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2

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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.1, ACCUMULATORS

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS SR 3.5.1.1 (ITS SR 3.5.1.1), ISTS SR 3.5.1.2 (ITS SR 3.5.1.2), ISTS SR 3.5.1.3 (ITS SR 3.5.1.3), ISTS SR 3.5.1.4 (ITS SR 3.5.1.4), and ISTS SR 3.5.1.5 (ITS SR 3.5.1.5) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a large break LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water. 1

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a large break LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the large break LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the large break LOCA. 1

BASES

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the large break LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. 1

The limiting large break LOCA is a double ended guillotine break ~~at the discharge of the reactor coolant pump~~ INSERT 1. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. 1

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators, safety injection pumps, and centrifugal charging pumps ~~both~~ each play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the safety injection and centrifugal charging pumps become ~~solely~~ responsible for terminating the temperature increase. 1

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a small break LOCA and there is a high probability that the criteria are met following a large break LOCA: 1

1 **INSERT 1**

Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase and the first few seconds of the refill phase of a large break LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46. 1 1

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve.~~ INSERT 2 The realistic large break LOCA safety analysis ~~assumes~~ takes values ~~of~~ between ~~[6468]~~ 7515 gallons and ~~[6879]~~ 8194 gallons. To allow for instrument inaccuracy, values of ~~[6520]~~ 7615 gallons and ~~[6820]~~ 7960 gallons are specified. INSERT 3 1 2 3

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

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1 **INSERT 2**

The large and small break LOCA safety analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models.

1 **INSERT 3**

The small break LOCA safety analysis assumes a value from within the range of values used for the large break safety analysis.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 4

The large and small break LOCA analyses are performed ~~at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. 1

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[~~2000~~]~~ psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. 2

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

1 **INSERT 4**

with accumulator pressures that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 600 psig and 683 psig. To allow for instrument inaccuracy, values of 624 psig and 668 psig are specified. The small break LOCA safety analysis assumes a value from the low end of the range of values taken for the large break safety analysis.

BASES

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, while current analysis techniques demonstrate that the accumulators ~~do not~~ discharge following a large main steam line break ~~for the majority of plants. Even if they do discharge~~, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

1

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 4).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

isolation

Each accumulator valve should be verified to be fully open. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. ~~[The Frequency of 12 hours is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.~~ 1

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

SR 3.5.1.2 and SR 3.5.1.3

Borated water volume and nitrogen cover pressure are verified for each accumulator. ~~[The Frequency of 12 hours is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.~~ 4

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

5

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. ~~[The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.~~

~~OR~~

4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
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~~--]~~

5

Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ ~~[~~2000~~]~~ psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. ~~[Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.~~ 2

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~. 1 2

2. 10 CFR 50.46.

3. FSAR, Chapter ~~[~~15~~]~~. 1 2

U

4. WCAP-15049-A, Rev. 1, April 1999.

5. NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a large break LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water. 1

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a large break LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the large break LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the large break LOCA. 1

BASES

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

large break

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. 1

The limiting large break LOCA is a double ended guillotine break ~~at the discharge of the reactor coolant pump~~. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. INSERT 1 1

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators, safety injection pumps, and centrifugal charging pumps ~~both~~ each play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the safety injection and centrifugal charging pumps become ~~solely~~ responsible for terminating the temperature increase. 1 1

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a small break LOCA and there is a high probability that the criteria are met following a large break LOCA: 1

1 **INSERT 1**

Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer.

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. Core is maintained in a coolable geometry. 1

Since the accumulators discharge during the blowdown phase and the first few seconds of the refill phase of a large break LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46. 1

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve.~~ INSERT 2 The realistic large break LOCA safety analysis ~~assumes~~ takes values ~~of~~ between ~~[6468]~~ 7515 gallons and ~~[6879]~~ 8194 gallons. To allow for instrument inaccuracy, values of ~~[6520]~~ 7615 gallons and ~~[6820]~~ 7960 gallons are specified. INSERT 3 1 2 3

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

1 **INSERT 2**

The large and small break LOCA safety analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models.

1 **INSERT 3**

The small break LOCA safety analysis assumes a value from within the range of values used for the large break safety analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 4

The large and small break LOCA analyses are performed ~~at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. 1

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[~~2000~~]~~ psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. 2

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

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1 **INSERT 4**

with accumulator pressures that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 600 psig and 683 psig. To allow for instrument inaccuracy, values of 624 psig and 668 psig are specified. The small break LOCA safety analysis assumes a value from the low end of the range of values taken for the large break safety analysis.

BASES

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, while current analysis techniques demonstrate that the accumulators ~~do not~~ discharge following a large main steam line break ~~for the majority of plants. Even if they do discharge~~, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits. 1

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 4).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

isolation

Each accumulator valve should be verified to be fully open. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. ~~[The Frequency of 12 hours is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.~~ 1

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 5

SR 3.5.1.2 and SR 3.5.1.3

Borated water volume and nitrogen cover pressure are verified for each accumulator. ~~[The Frequency of 12 hours is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.~~ 4

~~OR~~

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

5

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. ~~[The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.~~

~~OR~~

4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

5

Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ [2000] psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. ~~[Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

REFERENCES

1. FSAR, Chapter [6].

2. 10 CFR 50.46.

3. FSAR, Chapter [15].

4. WCAP-15049-A, Rev. 1, April 1999.

5. NUREG-1366, February 1990.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.1 BASES, ACCUMULATORS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS 3.5.1 Bases for the Applicable Safety Analysis have been changed to reflect Sequoyah (SQN) specific design. SQN does not give a specific value for instrument uncertainties in large break analysis

4. ISTS SR 3.5.1.1, ISTS SR 3.5.1.2, ISTS SR 3.5.1.3, ISTS SR 3.5.1.4 and ISTS SR 3.5.1.5 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.5.1.1, SR 3.5.1.2, SR 3.5.1.3, SR 3.5.1.4 and SR 3.5.1.5 is "In accordance with the Surveillance Frequency Control Program."

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.1, ACCUMULATORS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 2

ITS 3.5.2, ECCS - OPERATING

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

ITS A01 ITS 3.5.2

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.2 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

KNH-021

LCO 3.5.2 3.5.2 Two ECCS trains shall be OPERABLE.

STET ECCS L01

----------NOTES----------

LCO 3.5.2 Note 1 1. In MODE 3, both ~~safety injection (SI)~~ pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 4.4.6.3.

LCO 3.5.2 Note 2 2. In MODE 3, ECCS pumps may be made incapable of injecting to support transition into or from the APPLICABILITY of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds LTOP arming temperature ~~(350°F)~~ specified in the PTLR plus 25°F, whichever comes first.

---------- LA01

Applicability APPLICABILITY: MODES 1, 2 and 3.

ACTION:

ACTION A a. With one or more trains inoperable ~~and with at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available~~, restore the inoperable train(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. M01

ACTION B

ACTION C b. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, immediately enter LCO 3.0.3.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS train shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.1 a. ~~At least once per 12 hours~~ by verifying that the following valves are in the indicated positions with power to the valve operators removed:

ITS A01 ITS 3.5.2

EMERGENCY CORE COOLING SYSTEMS (ECCS)

SURVEILLANCE REQUIREMENTS (Continued)

	Valve Number	Valve Function	Valve Position
SR 3.5.2.1	a. FCV-63-1	RHR Suction from RWST	open
	b. FCV-63-22	SIS Discharge to Common Piping	open

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.2 SR 3.5.2.3 b. ~~At least once per 31 days~~ by:

SR 3.5.2.3 1. Verify ECCS piping is full of water ~~by venting the ECCS pump casings and accessible piping high points, and~~ LA03

SR 3.5.2.2 2. Verify each ECCS manual, power operated and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.

c. Deleted

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.8 d. ~~At least once per 18 months~~ perform a visual inspection of the containment sump and verify that the suction inlets are not restricted by debris and that the sump components (strainers, screens, etc.) show no evidence of structural distress or corrosion.

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.5 SR 3.5.2.6 e. ~~At least once per 18 months~~, by:

SR 3.5.2.5 1. Verifying that each automatic valve in the flow path that is not locked, sealed or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal.

SR 3.5.2.6 2. Verifying that each ECCS pump starts automatically on an actual or simulated actuation signal.

SR 3.5.2.4 f. By verifying that each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head when tested in accordance with the Inservice Testing Program of Specification 4.0.5.

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.7 g. ~~At least once per 18 months~~, verify the correct position of each mechanical stop for the following ECCS throttle valves:

Charging Pump Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
Valve Number	Valve Number	Valve Number
1. 63 - 582	1. 63 - 550	1. 63-542
2. 63 - 583	2. 63 - 552	2. 63-544
3. 63 - 584	3. 63 - 554	3. 63-546
4. 63 - 585	4. 63 - 556	4. 63-548

ITS   A01   ITS 3.5.2

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.2 — 3.5.2 Two ECCS trains shall be OPERABLE.

STET — ECCS — L01 — KNH-021

-----NOTES-----

LCO 3.5.2 Note 1 — 1. In MODE 3, both ~~safety injection (SI)~~ pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 4.4.6.3.

LCO 3.5.2 Note 2 — 2. In MODE 3, ECCS pumps may be made incapable of injecting to support transition into or from the APPLICABILITY of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds LTOP arming temperature ~~(350°F)~~ specified in the PTLR plus 25°F, whichever comes first.

----- LA01

Applicability — APPLICABILITY: MODES 1, 2 and 3.

ACTION:

ACTION A / ACTION B — a. With one or more trains inoperable ~~and with at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available~~, restore the inoperable train(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. M01

ACTION C — b. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, immediately enter LCO 3.0.3.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS train shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program — LA02

SR 3.5.2.1 — a. ~~At least once per 12 hours~~ by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. FCV-63-1	RHR Suction from RWST	open
b. FCV-63-22	SIS Discharge to Common Piping	open

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.2 SR 3.5.2.3 b. ~~At least once per 31 days~~ by:

SR 3.5.2.3 1. Verify ECCS piping is full of water ~~by venting the ECCS pump casings and accessible piping high points, and~~ LA03

SR 3.5.2.2 2. Verify each ECCS manual, power operated and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.

c. Deleted

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.8 d. ~~At least once per 18 months~~ perform a visual inspection of the containment sump and verify that the suction inlets are not restricted by debris and that the sump components (strainers, screens, etc.) show no evidence of structural distress or corrosion.

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.5 SR 3.5.2.6 e. ~~At least once per 18 months~~, by:

SR 3.5.2.5 1. Verifying that each automatic valve in the flow path that is not locked, sealed or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal.

SR 3.5.2.6 2. Verifying that each ECCS pump starts automatically on an actual or simulated actuation signal.

SR 3.5.2.4 f. By verifying that each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head when tested in accordance with the Inservice Testing Program of Specification 4.0.5.

In accordance with the Surveillance Frequency Control Program LA02

SR 3.5.2.7 g. ~~At least once per 18 months~~, verify the correct position of each mechanical stop for the following ECCS throttle valves:

Charging Pump Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
Valve Number	Valve Number	Valve Number
1. 63 - 582	1. 63 - 550	1. 63-542
2. 63 - 583	2. 63 - 552	2. 63-544
3. 63 - 584	3. 63 - 554	3. 63-546
4. 63 - 585	4. 63 - 556	4. 63-548

DISCUSSION OF CHANGES
ITS 3.5.2, ECCS - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 3.5.2 ACTION a entry condition is with one or more trains inoperable and with at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available. ITS 3.5.2 ACTION A entry condition is for one or more trains inoperable. This changes the CTS by deleting the requirement that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available to enter the required action for one or more inoperable ECCS train(s).

The purpose of CTS 3.5.2 ACTION a is to limit the time that the plant can continue to operate with one or more ECCS trains inoperable. Stating the entry condition for CTS 3.5.2 ACTION a as a compound condition could cause entry, exiting, and reentry into CTS 3.5.2 ACTION a based on whether the ECCS system has 100% flow capability available with one or more trains inoperable. This compound condition could incorrectly result in an inoperability time beyond 72 hours, without having returned a train to OPERABLE status. CTS 3.5.2 ACTION b Required Actions with less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available are retained in ITS LCO 3.5.2 ACTION C. By splitting CTS 3.5.2 ACTION a entry condition, where ITS LCO 3.5.2 ACTION A is entered when one or more ECCS trains are inoperable and ITS LCO 3.5.2 ACTION C is entered when less than 100% ECCS flow is available, ITS 3.5.2 ACTION A remains applicable regardless of overall remaining ECCS flow availability, so that the completion time clock is not reset in the event 100% flow is restored. This change is acceptable because ITS LCO 3.5.2 ACTION A and ACTION C provide adequate compensatory measures to take with one or more ECCS trains inoperable and address the condition of whether the ECCS system has at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available or not. This change is designated as more restrictive, because it limits the time the plant is allowed to operate with one or more ECCS train inoperable to 72 hours.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.5.2, ECCS - OPERATING

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.5.2 Note 2 allows an option in MODE 3 for the ECCS pumps to be made incapable of injecting to support transition into or from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds LTOP arming temperature (350°) specified in the PTLR plus 25°F. ITS LCO 3.5.2 Note 2 provides the same allowance but does not explicitly include the LTOP arming temperature of 350°F. This changes the CTS by moving the specific value of the LTOP arming temperature (350°) from the CTS to the PTLR.

The removal of this detail related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also this change is acceptable, because the removed information is adequately controlled in the PTLR. Changes to the PTLR are controlled in Chapter 5 of the Technical Specifications. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting the TS requirements is being removed from the ITS.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.5.2.a requires verification that the listed valves are in the indicated positions with power to the valve operators removed once per 12 hours. CTS 4.5.2.b.1 requires verification that the ECCS piping is full of water once per 31 days. CTS 4.5.2.b.2 requires verification that each ECCS manual, power operated and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position once per 31 days. CTS 4.5.2.d requires performance of a visual inspection of the containment sump, suction lines, and sump components once per 18 months. CTS 4.5.2.e.1 requires verification that each automatic valve in the flow path that is not locked, sealed or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal once per 18 months. CTS 4.5.2.e.2 requires verification that the ECCS pump starts automatically on an actual or simulated actuation signal once per 18 months. CTS 4.5.2.g requires verification that each mechanical stop of the ECCS throttle valves is in the correct position once per 18 months. ITS SR 3.5.2.1, ITS SR 3.5.2.2, ITS SR 3.5.2.3, ITS SR 3.5.2.5, ITS SR 3.5.2.6, ITS SR 3.5.2.7 and ITS SR 3.5.2.8 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the

DISCUSSION OF CHANGES
ITS 3.5.2, ECCS - OPERATING

control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 4.5.2.b.1 requires verifying ECCS piping is full of water by venting the ECCS pump casing and accessible piping high points. ITS SR 3.5.2.3 requires verifying ECCS piping is full of water. This changes the CTS by moving the details of how to vent the ECCS piping "by venting the ECCS pump casings and accessible piping high points" from the CTS to the Bases.

The removal of these details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable, because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

~~L01~~ Not Used ~~*(Category 1 – Relaxation of LCO Requirements)* CTS LCO 3.5.2 requires, two ECCS trains to be OPERABLE. CTS LCO 3.5.2 Note 1 states, in MODE 3, both safety injection SI pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 4.4.6.3. ITS LCO 3.5.2 requires two ECCS trains to be OPERABLE. ITS LCO 3.5.2 Note 1 states In MODE 3, both ECCS pump flow paths may be isolated for 2 hours to perform pressure isolation valve (PIV) testing per SR 3.4.14.1. This changes the CTS by allowing the RHR pump flow paths to be isolated in addition to the SI pump flow paths for Surveillance testing of the pressure isolation valves.~~ KNH-021

~~The purpose of CTS LCO 3.5.2 is to ensure that two ECCS trains are OPERABLE in MODES 1, 2, and 3. The purpose of ITS SR 3.4.14.1 is to prevent overpressure failure of the low pressure portions of connecting systems. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, or an unanalyzed accident that could degrade the ability for low pressure injection. PIV testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. Thus ITS SR 3.4.14.1 supports ITS LCO 3.5.2 to ensure that two ECCS trains are OPERABLE. Surveillance testing of the pressure isolation valves requires the SI Pump and RHR Pump flow paths to be isolated. CTS LCO 3.5.2 Note 1 allows~~

DISCUSSION OF CHANGES
ITS 3.5.2, ECCS - OPERATING

~~both SI pump flow paths to be isolated for two hours provided that the flow paths are readily restorable from the control room. In addition to isolating the SI pump flow paths, ITS LCO 3.5.2 Note 1 will allow both RHR pump flow paths to be isolated for two hours allowing for the required testing of the PIVs. This change permits the isolation of the ECCS (SI pump and RHR pump) flow paths provided that the flow paths are readily restorable from the control room. The acceptability of this testing allowance is based on the operability of the centrifugal charging system and the cold leg injection accumulators and the low probability of an accident occurring during the isolation time to support PIV testing. This change is acceptable because the LCO requirements continue to ensure that the system is maintained consistent with the safety analysis and licensing basis. This change is designated as less restrictive because the RHR pump flow path may be isolated to support testing of the PIVs.~~ KNH-021

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

KNH-021

4

3.5.2 LCO 3.5.2 Two ECCS trains shall be OPERABLE.

STET ECCS

---NOTES---

3.5.2 Note 1 ~~[~~ 1. In MODE 3, both ~~safety injection (SI)~~ pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. 1

3.5.2 Note 2 2. In MODE 3, ECCS pumps may be made incapable of injecting to support transition into or from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds ~~[375°F]~~ ~~[~~Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR plus ~~[~~25~~]~~°F~~]~~, whichever comes first. ~~]~~ 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a DOC M01	A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
ACTION a	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours
ACTION b	C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

KNH-021 3

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.5.2.a	SR 3.5.2.1	[Verify the following valves are in the listed position with power to the valve operator removed. INSERT 1 Number / Position / Function [ ] [ ] [ ] [ ] [ ] [ ] [ ] [ ] [ ]	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	2 1 2
4.5.2.b.2	SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	2 2
4.5.2.b.1	SR 3.5.2.3	[Verify ECCS piping is full of water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]]	2 2
4.5.2.f	SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	

1 **INSERT 1**

4.5.2.a	FCV-63-1	Open	RHR Suction from RWST
	FCV-63-22	Open	SIS Discharge to Common Piping

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CTS

ECCS - Operating
3.5.2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.e.1	SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.5.2.e.2	SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.5.2.g	SR 3.5.2.7	~~[~~ Verify, for each ECCS throttle valve listed below, each ~~position~~ mechanical stop is in the correct position. INSERT 2 ~~Valve Number~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	2 1 1 2

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1 **INSERT 2**

4.5.2.g

Charging Pump Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
63-582	63-550	63-542
63-583	63-552	63-544
63-584	63-554	63-546
63-585	63-556	63-548

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.d	SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

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CTS

ECCS - Operating
3.5.2

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

KNH-021

~~4~~

3.5.2 LCO 3.5.2 Two ECCS trains shall be OPERABLE.

STET

ECCS

-----NOTES-----

3.5.2 Note 1 ~~[~~ 1. In MODE 3, both ~~safety injection (SI)~~ pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. 1

3.5.2 Note 2 2. In MODE 3, ECCS pumps may be made incapable of injecting to support transition into or from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds ~~[375°F]~~ ~~[~~Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR plus ~~[~~25~~]~~°F~~]~~, whichever comes first. ~~]~~ 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a DOC M01	A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
ACTION a	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours
ACTION b	C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

KNH-021 3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.5.2.a	SR 3.5.2.1	[Verify the following valves are in the listed position with power to the valve operator removed. INSERT 1 Number / Position / Function [ ] [ ] [ ] [ ] [ ] [ ] [ ] [ ] [ ]	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	2 1 2
4.5.2.b.2	SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	2 2
4.5.2.b.1	SR 3.5.2.3	[Verify ECCS piping is full of water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]]	2 2
4.5.2.f	SR 3.5.2.4	Verify each ECCS pump's developed head at the test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	

1 INSERT 1

4.5.2.a	FCV-63-1	Open	RHR Suction from RWST
	FCV-63-22	Open	SIS Discharge to Common Piping

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CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.e.1	SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.5.2.e.2	SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.5.2.g	SR 3.5.2.7	~~[~~ Verify, for each ECCS throttle valve listed below, each ~~position~~ mechanical stop is in the correct position. INSERT 2 ~~Valve Number~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	2 1 1 2

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1 **INSERT 2**

4.5.2.g

Charging Pump Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
63-582	63-550	63-542
63-583	63-552	63-544
63-584	63-554	63-546
63-585	63-556	63-548

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.5.2.d SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

2

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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.2, ECCS - OPERATING

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS SR 3.5.2.1 (ITS SR 3.5.2.1), ISTS SR 3.5.2.2 (ITS SR 3.5.2.2), ISTS SR 3.5.2.3 (ITS SR 3.5.2.3), ISTS SR 3.5.2.5 (ITS SR 3.5.2.5), ISTS SR 3.5.2.6 (ITS SR 3.5.2.6), ISTS SR 3.5.2.7 (ITS SR 3.5.2.7) and ISTS SR 3.5.2.8 (ITS SR 3.5.2.8) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

3. ~~Editorial correction made for clarity.~~

~~4. Changes made to reflect changes made to the Specification.~~

Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

KNH-021

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,

b. Rod ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

INSERT 1

~~There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation.~~ In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 5.5 ~~24~~ hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation. 1

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

1 **INSERT 1**

There are two modes of ECCS operation, injection and recirculation. The injection mode consists of the injection phase and the recirculation mode consists of the cold leg recirculation phase and hot leg recirculation phase.

BASES

BACKGROUND (continued)

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, RHR heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines prior to entering the ~~boron injection tank (BIT) (if the plant utilizes a BIT)~~ centrifugal charging pump injection tank (CCPIT) and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. ~~Control~~ Throttle valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs. 1 1

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

The centrifugal charging and SI subsystems of the ECCS also function~~s~~ to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle. 1 6

BASES

BACKGROUND (continued)

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. ~~The actuation of safeguard loads is accomplished in a programmed time sequence.~~ If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA. 1

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

APPLICABLE SAFETY ANALYSES (continued)

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the required flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions: 1

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train ~~RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation)~~ (both containment spray trains are assumed to operate conservatively reducing containment pressure and increasing break flow) and 1

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling. 1

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either ECCS train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents. 1

BASES

LCO (continued)

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each ECCS train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring RHR suction to the containment sump. 1 1

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each ECCS train must maintain its designed independence to ensure that no single failure can disable both ECCS trains. 1

As indicated in Note 1, the SI ~~ECCS (SI Pump and RHR Pump)~~ STET flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. ~~7~~ KNH-021

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the LTOP Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the LTOP Applicability.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

BASES

APPLICABILITY (continued)

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 2

ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components. 1

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of ECCS trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable. 1

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

BASES

ACTIONS (continued)

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ~~or by key locking the control in the correct position~~ ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. ~~[A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.~~ 1 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. ~~[The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. Venting of the ECCS piping is accomplished by venting the pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. ~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~ 1 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for plants with flow limiting orifices.~~ ~~[The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.~~ 6 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

4

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.~~

~~OR~~

3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

4

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. U FSAR, Section ~~[ ]~~ 6.3.
4. U FSAR, Chapter ~~[~~15~~]~~, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.

1 5

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,

b. Rod ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

INSERT 1

~~There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation.~~ In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 5.5 ~~24~~ hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation. 1

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

1

INSERT 1

There are two modes of ECCS operation, injection and recirculation. The injection mode consists of the injection phase and the recirculation mode consists of the cold leg recirculation phase and hot leg recirculation phase.

BASES

BACKGROUND (continued)

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, RHR heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines prior to entering the ~~boron injection tank (BIT) (if the plant utilizes a BIT)~~ centrifugal charging pump injection tank (CCPIT) and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. ~~Control~~ Throttle valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs. 1 1

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

The centrifugal charging and SI subsystems of the ECCS also function~~s~~ to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle. 1 6

BASES

BACKGROUND (continued)

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. ~~The actuation of safeguard loads is accomplished in a programmed time sequence.~~ If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA. 1

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the required flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions: 1

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train ~~RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation)~~ (both containment spray trains are assumed to operate conservatively reducing containment pressure and increasing break flow) and 1

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling. 1

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either ECCS train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents. 1

BASES

LCO (continued)

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each ECCS train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the RHR containment sump. 1 1

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each ECCS train must maintain its designed independence to ensure that no single failure can disable both ECCS trains. 1

As indicated in Note 1, the ~~ECCS (SI Pump and RHR Pump)~~ SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. STET ~~7~~ KNH-021

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the LTOP Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the LTOP Applicability.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

BASES

APPLICABILITY (continued)

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 2

ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components. 1

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of ECCS trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable. 1

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

BASES

ACTIONS (continued)

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ~~or by key locking the control in the correct position~~ ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. ~~[A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.~~ 1 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. ~~[The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. Venting of the ECCS piping is accomplished by venting the pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. ~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~ 1 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 4

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for plants with flow limiting orifices.~~ 6 ~~[The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

4

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.~~

~~OR~~

3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~

4

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Section ~~[ ]~~ 6.3.
4. FSAR, Chapter ~~[~~15~~]~~, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.

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1 5

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.2 BASES, ECCS OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The listed LCOs concern the shutdown cooling function of the RHR System, not the ECCS function. The Applicability Section has adequately described why ECCS is not needed in MODES 5 and 6, and it is not necessary to describe why normal shutdown cooling is required. Therefore, this inappropriate information has been deleted.

3. ISTS SR 3.5.2.1, ISTS SR 3.5.2.2, ISTS SR 3.5.2.3, ISTS SR 3.5.2.5, ISTS SR 3.5.2.6, ISTS SR 3.5.2.7 and ISTS SR 3.5.2.8 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.5.2.1, SR 3.5.2.2, SR 3.5.2.3, SR 3.5.2.5, SR 3.5.2.6, SR 3.5.2.7 and SR 3.5.2.8 is "In accordance with the Surveillance Frequency Control Program."

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

6. Editorial changes made to enhance clarity/consistency.

KNH-021 ~~7. Changes are made to reflect changes made to the Specification.~~

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.2, ECCS - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.5.3, ECCS - SHUTDOWN

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

ITS A01 ITS 3.5.3

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.3 ECCS -SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.5.3 3.5.3 One ECCS train shall be OPERABLE.

LCO 3.5.3 Note 1

-----NOTE-----
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned to the ECCS mode of operation.

Applicability APPLICABILITY: MODE 4.

ACTION:

-----NOTE-----

LCO 3.5.3 ACTIONS Note

1. LCO 3.0.4b is not applicable to ECCS centrifugal charging subsystem.

LCO 3.5.3 Note 2

2. The required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves provided ~~that alternate heat removal methods are available via the steam generators to maintain reactor coolant system T_{avg} less than 350°F and provided~~ that the required subsystem is capable of being manually realigned to the ECCS mode of operation. A02

ACTION A a. With the required ECCS residual heat removal (RHR) subsystem inoperable, immediately initiate action to restore required ECCS RHR subsystem to OPERABLE status.

ACTION B b. With the required ECCS centrifugal charging subsystem inoperable, within one hour, restore required ECCS centrifugal charging subsystem to OPERABLE status, or be in COLD

ACTION C SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 4.5.3 The ECCS train shall be demonstrated OPERABLE per the following applicable Surveillance Requirements of 4.5.2:

SR 4.5.2.b.1
SR 4.5.2.d
SR 4.5.2.f
SR 4.5.2.g

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ITS

A01

ITS 3.5.3

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.4 DELETED

LIMITING CONDITION FOR OPERATION

This Specification is deleted.

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ITS

A01

ITS 3.5.3

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS -SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.5.3 3.5.3 One ECCS train shall be OPERABLE.

---NOTE---

LCO 3.5.3 Note 1 An RHR train may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned to the ECCS mode of operation.

Applicability APPLICABILITY: MODE 4.

ACTION:

---NOTE---

LCO 3.5.3 ACTIONS Note 1. LCO 3.0.4b is not applicable to ECCS centrifugal charging subsystem.

LCO 3.5.3 Note 2 2. The required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves ~~provided that alternate heat removal methods are available via the steam generators to maintain reactor coolant system~~ ~~T_{avg} less than 350°F and~~ provided that the required subsystem is capable of being manually realigned to the ECCS mode of operation. A02

ACTION A a. With the required ECCS residual heat removal (RHR) subsystem inoperable, immediately initiate action to restore required ECCS RHR subsystem to OPERABLE status.

ACTION B b. With the required ECCS centrifugal charging subsystem inoperable, within one hour, restore required ECCS centrifugal charging subsystem to OPERABLE status, or be in COLD

ACTION C SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 4.5.3 The ECCS train shall be demonstrated OPERABLE per the following applicable Surveillance Requirements of 4.5.2:

SR 4.5.2.b.1
SR 4.5.2.d
SR 4.5.2.f
SR 4.5.2.g

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 DELETED

LIMITING CONDITION FOR OPERATION

This Specification is deleted.

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DISCUSSION OF CHANGES
ITS 3.5.3, ECCS - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.5.3 ACTION Note 2 states, the required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves provided that alternate heat removal methods are available via the steam generators to maintain reactor coolant system T_{avg} less than 350° F and provided that the required subsystem is capable of being manually realigned to the ECCS mode of operation. ITS LCO 3.5.3 Note 2 states, the required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves provided the required subsystem is capable of being manually realigned to the ECCS mode of operation. This changes the CTS by removing the requirement for alternate heat removal methods from the ECCS Specification. ITS LCO 3.4.6 retains the requirements for decay heat removal in MODE 4.

This change is acceptable because the CTS requirements have not changed. ITS LCO 3.4.6 retains the requirements for decay heat removal in MODE 4. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

CTS

ECCS - Shutdown
3.5.3

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

3.5.3

LCO 3.5.3 One ECCS train shall be OPERABLE.

S

Residual Heat Removal (RHR)

4

---NOTE---

3.5.3 Note 1

1. An ~~RHR~~ train may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned to the ECCS mode of operation. 2 4

INSERT 1 2

Applicability

APPLICABILITY: MODE 4.

ACTIONS

centrifugal charging 1

---NOTE---

ACTION Note

LCO 3.0.4.b is not applicable to ECCS ~~high head~~ subsystem.

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. [Required ECCS ~~residual heat removal~~ ~~(~~RHR~~)~~ subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately]	3
ACTION b	B. Required ECCS ~~[high head~~ centrifugal charging subsystem] inoperable.	B.1 Restore required ECCS ~~[high head~~ centrifugal charging subsystem] to OPERABLE status.	1 hour	3
ACTION b	C. Required Action and associated Completion Time [of Condition B] not met.	C.1 Be in MODE 5.	24 hours	3

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Amendment XXX

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2 INSERT 1

3.5.3 Note 2

2. The required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves provided that the required subsystem is capable of being manually realigned to the ECCS mode of operation.

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SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.5.3	SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE:	In accordance with applicable SRs
4.5.2.b.1 4.5.2.d 4.5.2.f 4.5.2.g		~~[SR 3.5.2.1]~~ ~~[~~SR 3.5.2.7~~]~~ ~~[~~SR 3.5.2.3~~]~~ SR 3.5.2.8 SR 3.5.2.4	

3

CTS

ECCS - Shutdown
3.5.3

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

3.5.3 LCO 3.5.3 One ECCS train shall be OPERABLE.

Residual Heat Removal (RHR) S 4

-------------------NOTE-------------------

3.5.3 Note 1 1. An ~~RHR~~ train may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned to the ECCS mode of operation. 2 4

INSERT 1 2

Applicability APPLICABILITY: MODE 4.

ACTIONS

centrifugal charging 1

-------------------NOTE-------------------

ACTION Note LCO 3.0.4.b is not applicable to ECCS ~~high head~~ subsystem.

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION a	A. [Required ECCS ~~residual heat removal~~ (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately]	3
ACTION b	B. Required ECCS ~~[high head~~ centrifugal charging subsystem] inoperable.	B.1 Restore required ECCS ~~[high head~~ centrifugal charging subsystem] to OPERABLE status.	1 hour	3
ACTION b	C. Required Action and associated Completion Time [of Condition B] not met.	C.1 Be in MODE 5.	24 hours	3

2 INSERT 1

3.5.3 Note 2

2. The required ECCS residual heat removal (RHR) subsystem may be inoperable for up to 1 hour for surveillance testing of valves provided that the required subsystem is capable of being manually realigned to the ECCS mode of operation.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.5.3	SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE: ~~[SR 3.5.2.1]~~ [SR 3.5.2.7] [SR 3.5.2.3] SR 3.5.2.8 SR 3.5.2.4	In accordance with applicable SRs

4.5.2.b.1
4.5.2.d
4.5.2.f
4.5.2.g

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.3, ECCS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description

2. LCO 3.5.3 Note 2 has been added to be consistent with Sequoyah current licensing basis. Note 2 allows the required ECCS residual heat removal (RHR) subsystem to be inoperable for up to 1 hour for surveillance testing of valves provided the required subsystem is capable of being manually realigned to the ECCS mode of operation (CTS 3.5.3 Action Note 2).

3. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. Editorial correction made for clarity.

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

RHR

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2. 1

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that ~~certain~~ automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. 1

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

and adequate core cooling is maintained

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

BASES

LCO (continued)

two Notes. Note 1

This LCO is modified by ~~a Note~~ ~~that~~ allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4. 8

INSERT 1 8

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 2

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS ~~high head~~ subsystem when entering MODE 4. There is an increased risk 1

centrifugal charging

associated with entering MODE 4 from MODE 5 with an inoperable ECCS ~~high head~~ subsystem and the provisions of LCO 3.0.4.b, which allow 1

entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. ~~Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.~~ 3

1

INSERT 1

A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring manual realignment capability (FCV-74-1 and FCV-74-2 can be opened from the main control room) ensures that in the unlikely event of a DBA during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event.

BASES

ACTIONS (continued)

subsystems

With both RHR ~~pumps and heat exchangers~~ inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status. 4

B.1

centrifugal charging

With no ECCS ~~high head~~ subsystem OPERABLE, ~~due to the inoperability of the centrifugal charging pump or flow path from the RWST,~~ the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging ~~high head~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity ~~or to initiate actions to place the plant in MODE 5, where an ECCS train is not required~~. 1 5 1 6

C.1

the plant should be placed in MODE 5

When the Required Actions of Condition B cannot be completed within the required Completion Time, ~~a controlled shutdown should be initiated~~. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators. 7

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND	The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications. In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head). The ECCS flow paths consist of piping, valves, RHR heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2. 1
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section. Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that ~~certain~~ automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. 1 Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core and adequate core cooling is maintained following a DBA. In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump. During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

BASES

LCO (continued)

two Notes. Note 1

This LCO is modified by ~~a Note that~~ allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. INSERT 1 This allows operation in the RHR mode during MODE 4. 8 8

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 2

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS ~~high head~~ centrifugal charging subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS ~~high head~~ centrifugal charging subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance. 1 1

<u>A.1</u>

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. ~~Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.~~ 3

1 **INSERT 1**

A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring manual realignment capability (FCV-74-1 and FCV-74-2 can be opened from the main control room) ensures that in the unlikely event of a DBA during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event.

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BASES

ACTIONS (continued)

subsystems

With both RHR ~~pumps and heat exchangers~~ inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status. 4

B.1

centrifugal charging

With no ECCS ~~high head~~ subsystem OPERABLE, ~~due to the inoperability of the centrifugal charging pump or flow path from the RWST,~~ the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging ~~high head~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity ~~or to initiate actions to place the plant in MODE 5, where an ECCS train is not required~~. 1 5 1 6

C.1

the plant should be placed in MODE 5

When the Required Actions of Condition B cannot be completed within the required Completion Time, ~~a controlled shutdown should be initiated~~. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators. 7

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.3 BASES, ECCS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The MODE 5 and MODE 6 core cooling LCOs listed are relative to the shutdown cooling function of the RHR System and not the ECCS function. The Applicability Section has adequately described why ECCS is not needed in MODES 5 and 6, and it is not necessary to address the requirements of the shutdown cooling function. Therefore, this inappropriate information has been deleted.

3. The statement in ACTION A.1 Bases concerning how decay heat is removed is not appropriate for this Specification since ITS 3.5.3 is relative to ECCS and not decay heat removal. Normal decay heat removal in MODE 4 is addressed in ITS LCO 3.4.6. In addition, Required Action A.1 of the Specification addresses the requirements to restore the ECCS RHR subsystem for ECCS purposes and not normal decay heat removal. Therefore, the statements discussing decay heat removal have been deleted.

4. ISTS 3.5.3 ACTION A.1 Bases states "With both RHR pumps and heat exchangers inoperable..." ITS 3.5.3 ACTION A.1 Bases states "With both RHR subsystems inoperable..." This changes the ISTS 3.5.3 ACTION A.1 Bases by expanding the reasons that a RHR subsystem may be inoperable beyond a pump and/or heat exchanger being inoperable. This is acceptable, since there may be other reasons that both RHR subsystems are inoperable, and the statement that both RHR subsystems are inoperable is sufficient and is consistent with the actual wording of ITS Required Action A.1. In addition, the required components of an OPERABLE RHR subsystem, including pumps and heat exchangers, are defined in other sections of the ITS 3.5.3 Bases, including the third paragraph of the Background section, and the second paragraph of the LCO section.

5. ISTS 3.5.3 ACTION B.1 Bases states "With no ECCS high head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path of the RWST..." ITS 3.5.3 ACTION B.1 Bases states "With no ECCS centrifugal charging subsystem OPERABLE..." This changes the ISTS 3.5.3 ACTION B.1 Bases by deleting the statement concerning how a centrifugal charging subsystem is determined to be inoperable. This is acceptable, since there may be other reasons that the ECCS centrifugal charging subsystem is inoperable, and the statement that the ECCS centrifugal charging subsystem is inoperable is sufficient and is consistent with the actual wording of ITS Required Action B.1. In addition, the required components of an OPERABLE centrifugal charging subsystem, including pumps and suction source, are defined in other sections of the ITS 3.5.3 Bases, including the third paragraph of the Background section, and the second paragraph of the LCO section.

6. The statement in ACTION B.1 Bases regarding initiation of actions to place the plant in MODE 5 has been deleted, since the statement is not consistent with the actual wording of ITS Required Action B.1. ITS Required Action B.1 does not address a plant cooldown to MODE 5; it only addresses restoring the inoperable ECCS subsystem to OPERABLE status. ITS Required Action C.1 contains the requirement to place the unit in MODE 5.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.3 BASES, ECCS - SHUTDOWN

7. ISTS ACTION C.1 states, in part, that a controlled shutdown should be initiated when the Required Actions of Condition B cannot be completed within the required Completion Time. ITS ACTION C.1, states, in part, that the plant should be placed in MODE 5 if the Required Actions of Condition B cannot be completed within the required Completion Time. This change is acceptable since the statement is consistent with the actual wording of ITS Required Action C.1 and is a more accurate action statement than the ISTS Bases statement that a controlled shutdown should be initiated.
8. Changes have been made to reflect changes made to Specification.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.3, ECCS - SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 4

ITS 3.5.4, REFUELING WATER STORAGE TANK

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

ITS A01 ITS 3.5.4

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

LCO 3.5.4 3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

SR 3.5.4.2 a. A contained borated water volume of ~~between~~ greater than or equal to 370,000 ~~and 375,000~~ gallons, LA02 A02

SR 3.5.4.3 b. A boron concentration of between 2500 and 2700 ppm of boron,

SR 3.5.4.1 c. A minimum solution temperature of 60°F, and

SR 3.5.4.1 d. A maximum solution temperature of 105°F.

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

ACTION: Add proposed Action A L01

ACTION B With the RWST inoperable, for reasons other than Condition A restore the tank to OPERABLE status within 1 hour or be in at least HOT
ACTION C STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

SR 3.5.4.2
SR 3.5.4.3 a. ~~At least once per 7 days~~ In accordance with the Surveillance Frequency Control Program by: LA01

SR 3.5.4.2 1. Verifying the contained borated water volume in the tank, and

SR 3.5.4.3 2. Verifying the boron concentration of the water.

SR 3.5.4.1 b. ~~At least once per 24 hours~~ In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature. LA01

ITS

A01

ITS 3.5.4

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

LCO 3.5.4 3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

LA02

SR 3.5.4.2 a. A contained borated water volume of ~~between~~ 370,000 ~~and 375,000~~ gallons,

greater than or equal to A02

SR 3.5.4.3 b. A boron concentration of between 2500 and 2700 ppm of boron,

SR 3.5.4.1 c. A minimum solution temperature of 60°F, and

SR 3.5.4.1 d. A maximum solution temperature of 105°F.

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

Add proposed Action A L01

ACTION:

for reasons other than Condition A

ACTION B With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT
ACTION C STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program LA01

SR 3.5.4.2 SR 3.5.4.3 a. ~~At least once per 7 days~~ by:

SR 3.5.4.2 1. Verifying the contained borated water volume in the tank, and

SR 3.5.4.3 2. Verifying the boron concentration of the water.

In accordance with the Surveillance Frequency Control Program LA01

SR 3.5.4.1 b. ~~At least once per 24 hours~~ by verifying the RWST temperature.

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DISCUSSION OF CHANGES
ITS 3.5.4, REFUELING WATER STORAGE TANK

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 3.5.5 a requires, in part, the refueling water storage tank (RWST) shall be OPERABLE with a contained borated water volume of between 370,000 and 375,000 gallons. ITS SR 3.5.4.2 requires, in part, a similar requirement of the RWST volume of greater than or equal to 370,000 gallons. This changes the CTS by explicitly requiring the RWST minimum volume to be greater than or equal to 370,000 gallons. The discussion for removing the maximum RWST borated volume of 375,000 gallons is contained in DOC LA02.

This change is acceptable because no changes are made to the CTS requirements. The change in format from the CTS to the ITS maintains the technical requirements of the minimum required RWST level. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.5.5.a.1 requires verifying the contained borated water volume in the RWST tank once per 7 days. CTS 4.5.5.a.2 requires verifying the boron concentration in the RWST once per 7 days. CTS 4.5.5.b requires the verification of the RWST temperature once per 24 hours. ITS SR 3.5.4.1, SR 3.5.4.2, and SR 3.5.4.3 require similar Surveillances and specify the periodic Frequency as, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequencies for these SRs and associated Bases to the Surveillance Frequency Control Program.

DISCUSSION OF CHANGES
ITS 3.5.4, REFUELING WATER STORAGE TANK

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequencies are removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 3.5.5 a requires the refueling water storage tank (RWST) to be OPERABLE with a contained borated water volume of between 370,000 and 375,000 gallons. ITS SR 3.5.4.2 requires verification of RWST borated water volume is greater than or equal to 370,000 gallons. This changes the CTS by moving the stated maximum RWST borated water volume of 375,000 to the SQN UFSAR.

The removal of these details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to verify the minimum required RWST borated water volume. This change is acceptable because the UFSAR contains the maximum borated water volume for the RWST. The removed requirements will be adequately controlled in the UFSAR as any changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.5.5 ACTION allows 1 hour to restore an inoperable RWST. ITS 3.5.4 ACTION A allows 8 hours to restore the RWST to OPERABLE status if the inoperability is due to the RWST boron concentration or temperature not within limits. ITS 3.5.4 ACTION B requires the restoration of the RWST to an OPERABLE status within 1 hour for reasons other than Condition A. This changes the CTS by increasing the Completion Time for restoration of an inoperable RWST due to boron concentration or temperature not within limits from 1 hour to 8 hours.

The purpose of CTS 3.5.5 Action is to require rapid correction of conditions that affect both trains of ECCS. This change is acceptable because the Completion

DISCUSSION OF CHANGES
ITS 3.5.4, REFUELING WATER STORAGE TANK

Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The primary function of the RWST is to provide large volumes of water to the RCS following a Loss of Coolant Accident. This large volume of water continues to be available while in this Condition. As a result, the most important safety function of the RWST can still be provided. Because of the volume of the RWST, changes to the boron concentration or temperature occur slowly, and consequently would not go far out of limit. If one of these parameters was out of limit, more than one hour would likely be required to restore the parameter. Given the remaining abilities of the RWST, requiring a plant shutdown after one hour is not warranted. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

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

CTS

RWST
3.5.4

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

3.5.5 LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01	A. RWST boron concentration not within limits. <u>OR</u> RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
ACTION	B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
ACTION	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

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CTS

RWST
3.5.4

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
3.5.5.c 3.5.5.d 4.5.5.b	SR 3.5.4.1	~~-----NOTE-----~~ ~~[Only required to be performed when ambient air temperature is < [35]°F or > [100]°F.]~~ Verify RWST borated water temperature is ≥ ~~[35]~~ 60 °F and ≤ ~~[100]~~ 105 °F.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 3 2 3
3.5.5.a 4.5.5.a.1	SR 3.5.4.2	Verify RWST borated water volume is ≥ ~~[466,200~~ 370,000 gallons ~~( )%]~~.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 2 3
3.5.5.b 4.5.5.a.2	SR 3.5.4.3	Verify RWST boron concentration is ≥ ~~[2000]~~ 2500 ppm and ≤ ~~[2200]~~ 2700 ppm.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 2 3

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

3.5.5 LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01	A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
ACTION	B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
ACTION	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

CTS

RWST
3.5.4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
3.5.5.c 3.5.5.d 4.5.5.b SR 3.5.4.1	~~-----NOTE----- [Only required to be performed when ambient air temperature is < [35]°F or > [100]°F.] -----~~ (1) Verify RWST borated water temperature is ≥ ~~[35]~~ 60 °F and ≤ ~~[100]~~ 105 °F.	~~[24 hours~~ ~~OR~~ (3) (2) In accordance with the Surveillance Frequency Control Program ~~]~~ (3)
3.5.5.a 4.5.5.a.1 SR 3.5.4.2	Verify RWST borated water volume is ≥ ~~[466,200~~ 370,000 gallons ~~( )%]~~.	~~[7 days~~ ~~OR~~ (3) (2) In accordance with the Surveillance Frequency Control Program ~~]~~ (3)
3.5.5.b 4.5.5.a.2 SR 3.5.4.3	Verify RWST boron concentration is ≥ ~~[2000]~~ 2500 ppm and ≤ ~~[2200]~~ 2700 ppm.	~~[7 days~~ ~~OR~~ (3) (2) In accordance with the Surveillance Frequency Control Program ~~]~~ (3)

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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.4, REFUELING WATER STORAGE TANKS (RWST)

1. A bracketed Note for ISTS SR 3.5.4.1 associated with the effect of ambient air temperature on RWST temperature is not adopted. SQN RWST borated water is heated and not maintained at ambient temperature, and the current temperature band is not very large.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS SR 3.5.4.1 (ITS SR 3.5.4.1), ISTS SR 3.5.4.2 (ITS SR 3.5.4.2), and ISTS SR 3.5.4.3 (ITS SR 3.5.4.3) provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies for these SRs under the Surveillance Frequency Control Program.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling cavity ~~pool~~ during refueling, and to the ECCS and the Containment Spray System during accident conditions. 1

The RWST supplies both trains of the ECCS and the Containment Spray System through a common ~~separate, redundant~~ supply header~~s~~ during the injection phase of a loss of coolant accident (LOCA) recovery. ~~A~~ M~~m~~otor operated isolation valves ~~is~~ are provided in the ~~each~~ header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. 1 The recirculation mode is entered when ECCS pump suction is transferred to the containment sump following receipt of the RWST - Low ~~Low (~~Level ~~1)~~ signal INSERT 1. INSERT 2 Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events. 1

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. ~~Each set of~~ The isolation valves ~~is~~ are interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases. 1

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

1 **INSERT 1**

coincident with Containment Sump Level – High signal

1 **INSERT 2**

The transfer of the containment spray pump suction to the containment sump is manually initiated upon receipt of a high level in the containment sump or the RWST Low-Low (Level) alarm.

BASES

BACKGROUND (continued)

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase~~,~~ ; 2

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling, and ; 2

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray ~~and Cooling~~ System~~s~~." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses. 1

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. ~~The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown~~ 3

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations~~. The maximum RWST temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum RWST temperature is an assumption in ~~both~~ the MSLB analysis ~~and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting~~. 3 1

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 28 ~~[27]~~ seconds, with offsite power available, or 58 ~~[37]~~ seconds without offsite power. ~~This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves, and a [10] second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~ 4 1 3

1

For a large break LOCA analysis, the minimum water volume limit of 370,000 ~~[466,200]~~ gallons and the lower boron concentration limit of 2500 ~~[2000]~~ ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core. 4

The upper limit on boron concentration of 2700 ~~[2200]~~ ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg recirculation ~~injection~~ is to minimize the potential for ~~avoid~~ boron precipitation in the core following the accident. 4 1

~~In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of [35]°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of [100]°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower~~ 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.~~ 1

The RWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 5

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST ~~tank~~ to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection. 1

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

In this ~~C~~c Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST ~~tank~~ to OPERABLE status ~~or to place the plant in a MODE in which the RWST is not required~~. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains. 9 1

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band. ~~[The Frequency of 24 hours is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 7

~~The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.~~ 8

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. ~~[Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 7

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~[Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--~~ ~~]~~

7

REFERENCES U 1. FSAR, Chapter ~~[~~6~~]~~ and Chapter ~~[~~15~~]~~. 1 4

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling cavity ~~pool~~ during refueling, and to the ECCS and the Containment Spray System during accident conditions. 1

The RWST supplies both trains of the ECCS and the Containment Spray System through a common ~~separate, redundant~~ supply header~~s~~ during the injection phase of a loss of coolant accident (LOCA) recovery. ~~A m~~ M otor operated isolation valve s ~~is~~ are provided in ~~each~~ the header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. 1 The recirculation mode is entered when ECCS pump suction is transferred to the containment sump following receipt of the RWST - Low ~~Low (~~Level ~~1)~~ signal INSERT 1. INSERT 2 Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events. 1

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. The ~~Each set of~~ isolation valves ~~is~~ are interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases. 1

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

1

INSERT 1

coincident with Containment Sump Level – High signal

1

INSERT 2

The transfer of the containment spray pump suction to the containment sump is manually initiated upon receipt of a high level in the containment sump or the RWST Low-Low (Level) alarm.

BASES

BACKGROUND (continued)

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase~~,~~ ; 2

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling, and ; 2

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray ~~and Cooling~~ System~~s~~." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses. 1

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. ~~The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown~~ 3

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations~~. The maximum RWST temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum RWST temperature is an assumption in ~~both~~ the MSLB analysis ~~and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting~~. 3 1

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as ~~[27]~~ 28 seconds, with offsite power available, or ~~[37]~~ 58 seconds without offsite power. ~~This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves, and a [10] second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~ 4 1 3

1

For a large break LOCA analysis, the minimum water volume limit of ~~[466,200]~~ 370,000 gallons and the lower boron concentration limit of ~~[2000]~~ 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core. 4

The upper limit on boron concentration of ~~[2200]~~ 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg ~~injection~~ recirculation is to ~~avoid~~ minimize the potential for boron precipitation in the core following the accident. 4 1

~~In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of [35]°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of [100]°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower~~ 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.~~ 1

The RWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. ~~Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."~~ 5

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST ~~tank~~ to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection. 1

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

c

In this ~~C~~ondition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST ~~tank~~ to OPERABLE status ~~or to place the plant in a MODE in which the RWST is not required~~. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains. 9 1

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band. ~~[The Frequency of 24 hours is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~ 7

~~The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.~~ 8

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. ~~[Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 7

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~[Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.~~ 6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

7

REFERENCES U 1. FSAR, Chapter ~~[~~6~~]~~ and Chapter ~~[~~15~~]~~. 1 4

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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.4 BASES, REFUELING WATER STORAGE TANK (RWST)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.
3. ISTS Bases 3.5.4 Applicable Safety Analyses section, second paragraph, includes the phrase "The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements..." ISTS Bases 3.5.4 Applicable Safety Analyses section, third paragraph, includes the sentence "Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core." ITS Bases 3.5.4 does not include this phrase and sentence. This is acceptable, because the plant specific design includes a BIT, but there are no minimum boron concentration design and licensing basis requirements for the BIT. Therefore, deletion of this information is consistent with the current plant design and licensing basis.
4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.
5. The MODE 5 and MODE 6 core cooling LCOs listed are relative to the shutdown cooling function of the RHR System and not the ECCS function. The Applicability Section has adequately described why ECCS is not needed in MODES 5 and 6, and it is not necessary to address the requirements of the shutdown cooling function. Therefore, this inappropriate information has been deleted.
6. ISTS SR 3.5.4.1, ISTS SR 3.5.4.2, and ITS SR 3.5.4.3 provide two options for controlling the Frequencies of Surveillance Requirements. SQN is proposing to control the Surveillance Frequencies under the Surveillance Frequency Control Program. Therefore, the Frequency for ITS SR 3.5.4.1, SR 3.5.4.2, and SR 3.5.4.3 is "In accordance with the Surveillance Frequency Control Program."
7. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.
8. Changes are made to reflect those changes made to the Specification. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.
9. Typographical/grammatical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.4, REFUELING WATER STORAGE TANK

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 5

ITS 3.5.5, SEAL INJECTION FLOW

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

ITS

A01

ITS 3.5.5

COOLING SYSTEMS (ECCS)

3/4.5.6 SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

LCO 3.5.5

3.5.6 Reactor coolant pump seal injection flow shall be within the limits of Figure 3.5.6-1.

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

ACTION A

ACTION B

With reactor coolant pump seal injection flow not within limits, adjust manual seal injection throttle valves to give a flow within limit in accordance with Surveillance Requirement 4.5.6 within 4 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

In accordance with the Surveillance Frequency Control Program

LA01

SR 3.5.5.1

4.5.6 ~~At least once per 31 days~~* verify manual seal injection throttle valves are adjusted to give a flow within the emergency core cooling system safety analysis limits in Figure 3.5.6-1.

SR 3.5.5.1 Note

*This surveillance is not required to be performed until 4 hours after the reactor coolant system pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig.

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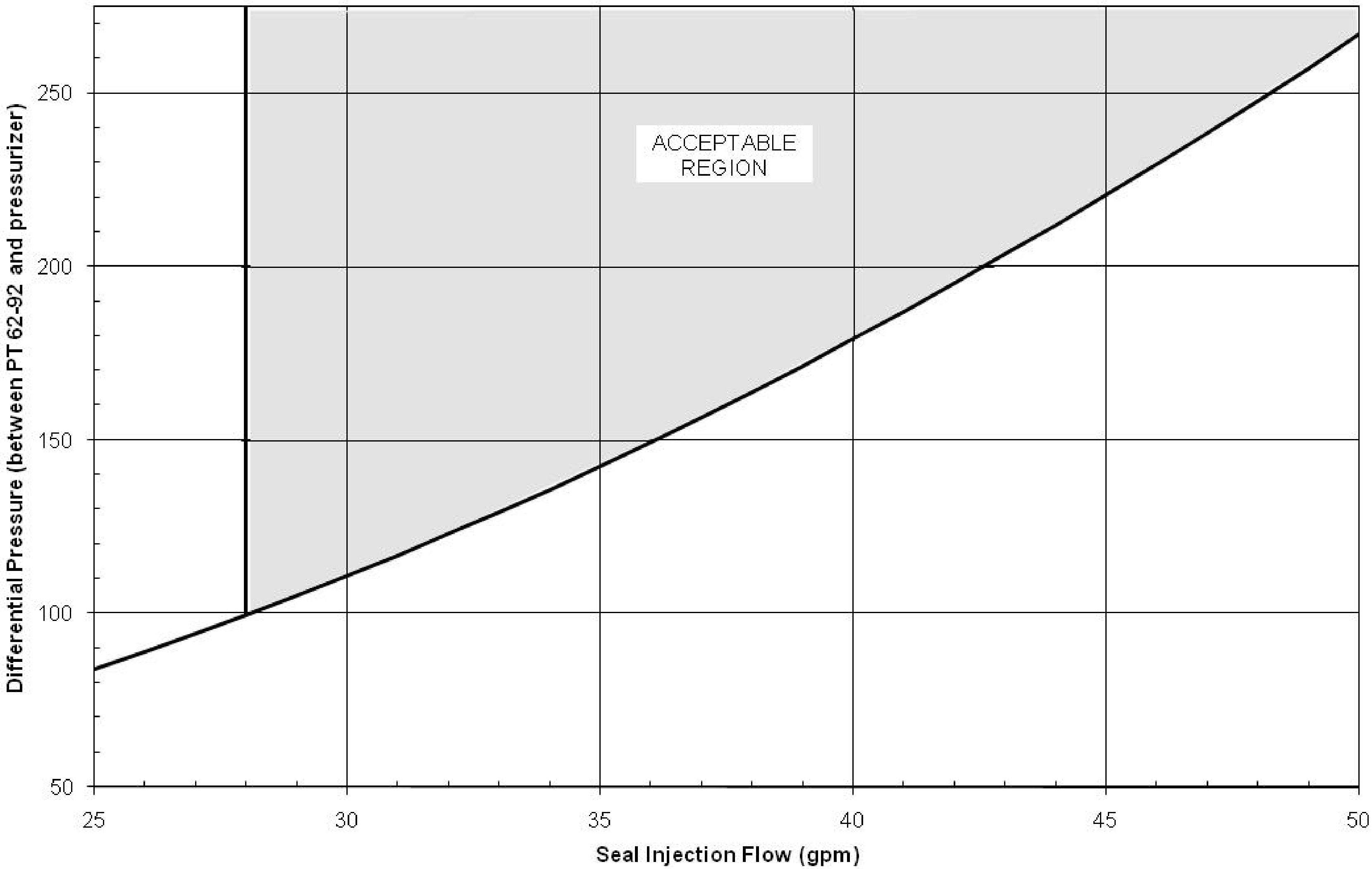
A01

FIGURE 3.5.6-1

Seal Injection Flow Limits

Figure 3.5.5-1



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EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.6 SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

LCO 3.5.5 3.5.6 Reactor coolant pump seal injection flow shall be within the limits of Figure 3.5.6-1.

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTION:

ACTION A — With reactor coolant pump seal injection flow not within limits, adjust manual seal injection throttle valves to give a flow within limit in accordance with Surveillance Requirement 4.5.6 within 4 hours. ACTION B — Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

In accordance with the Surveillance Frequency Control Program LA01

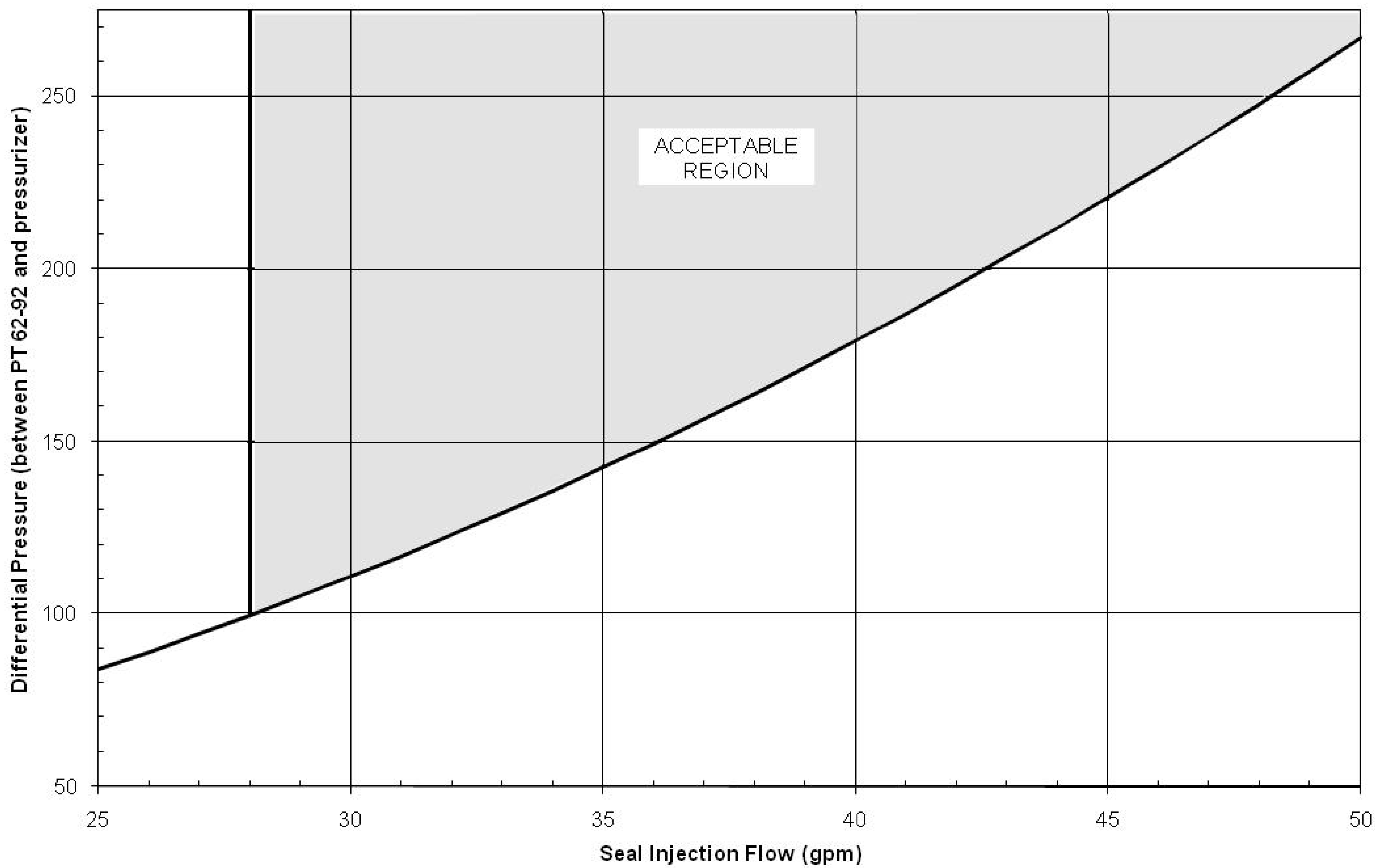
SR 3.5.5.1 4.5.6 ~~At least once per 31 days~~* verify manual seal injection throttle valves are adjusted to give a flow within the emergency core cooling system safety analysis limits in Figure 3.5.6-1.

SR 3.5.5.1 Note * This surveillance is not required to be performed until 4 hours after the reactor coolant system pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig.

FIGURE 3.5.6-1

Seal Injection Flow Limits

Figure 3.5.5-1



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DISCUSSION OF CHANGES
ITS 3.5.5, SEAL INJECTION FLOW

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 4.5.6 requires, in part, verifying manual seal injection throttle valve are adjusted to give a flow within the limits of Figure 3.5.6-1 every 31 days. ITS SR 3.5.5.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified Frequency for this SR and associated Bases to the Surveillance Frequency Control Program.

The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. A new program (Surveillance Frequency Control Program) is being added to the Administrative Controls section of the Technical Specifications describing the control of Surveillance Frequencies. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies will be in accordance with the Surveillance Frequency Control Program. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
ITS 3.5.5, SEAL INJECTION FLOW

LESS RESTRICTIVE CHANGES

None

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

3.5.6 LCO 3.5.5 Reactor coolant pump seal injection flow ~~[resistance]~~ shall be ~~[≤ [40] gpm with [centrifugal charging pump discharge header] pressure ≥ [2480] psig and the [charging flow] control valve full open or ≥ [0.2117] ft/gpm² or~~ within the limits of Figure 3.5.5-1~~]~~. 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION	A. Seal injection flow ~~[resistance]~~ not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow ~~[resistance]~~ within limit.	4 hours
ACTION	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

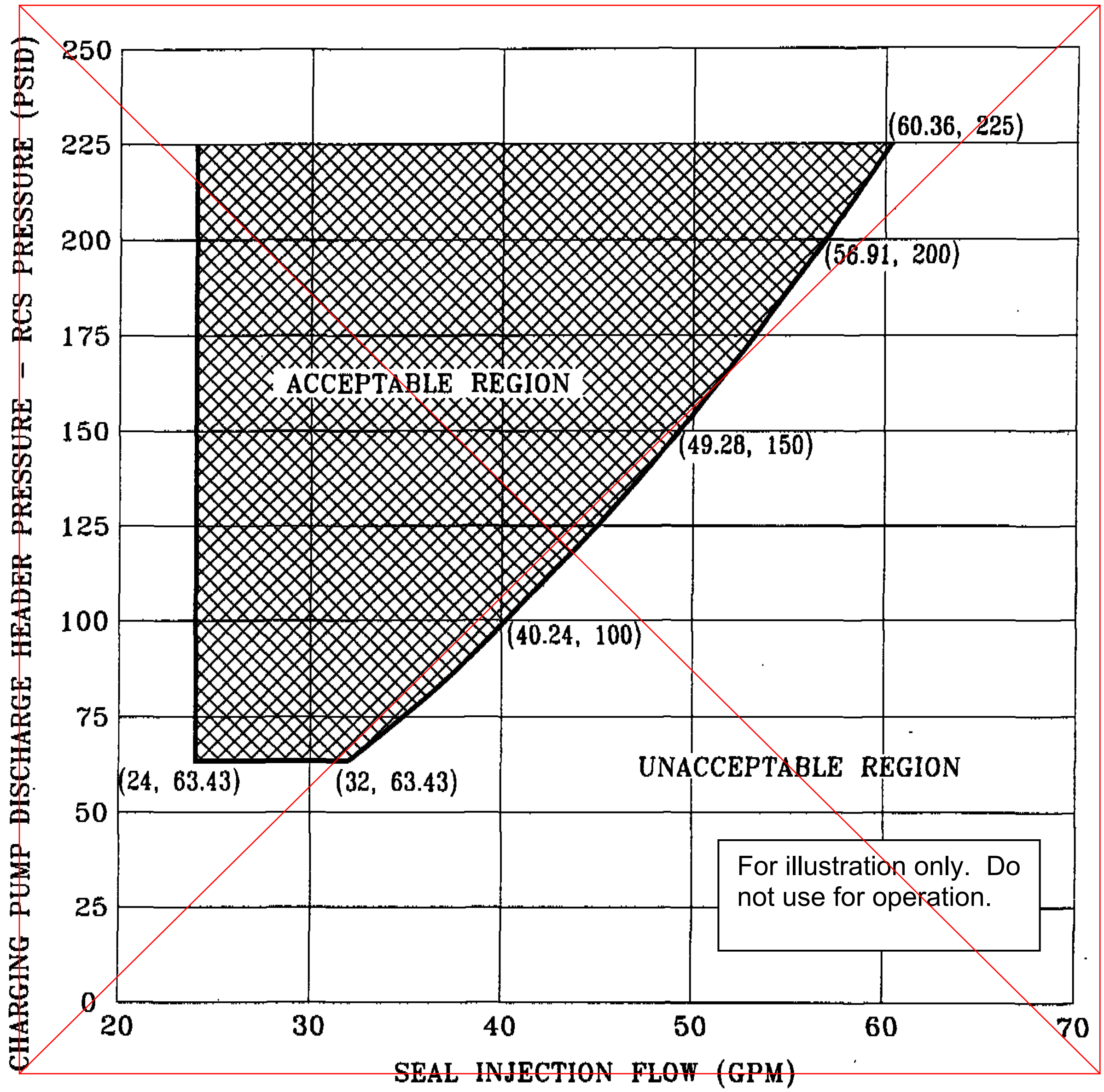
1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
4.5.6 * SR 3.5.5.1	------------------------------NOTE------------------------------ Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at ≥ ~~[~~2215 psig and ≤ 2255 psig~~]~~. --		1
4.5.6	Verify manual seal injection throttle valves are adjusted to give a flow ~~[resistance] [of ≤ [40 gpm] with [centrifugal charging pump discharge header] pressure ≥ [2480] psig and the [charging flow] control valve full open or ≥ [0.2117] ft/gpm^2 or~~ within the limit of Figure 3.5.5-1.~~]~~	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

CTS

INSERT 1



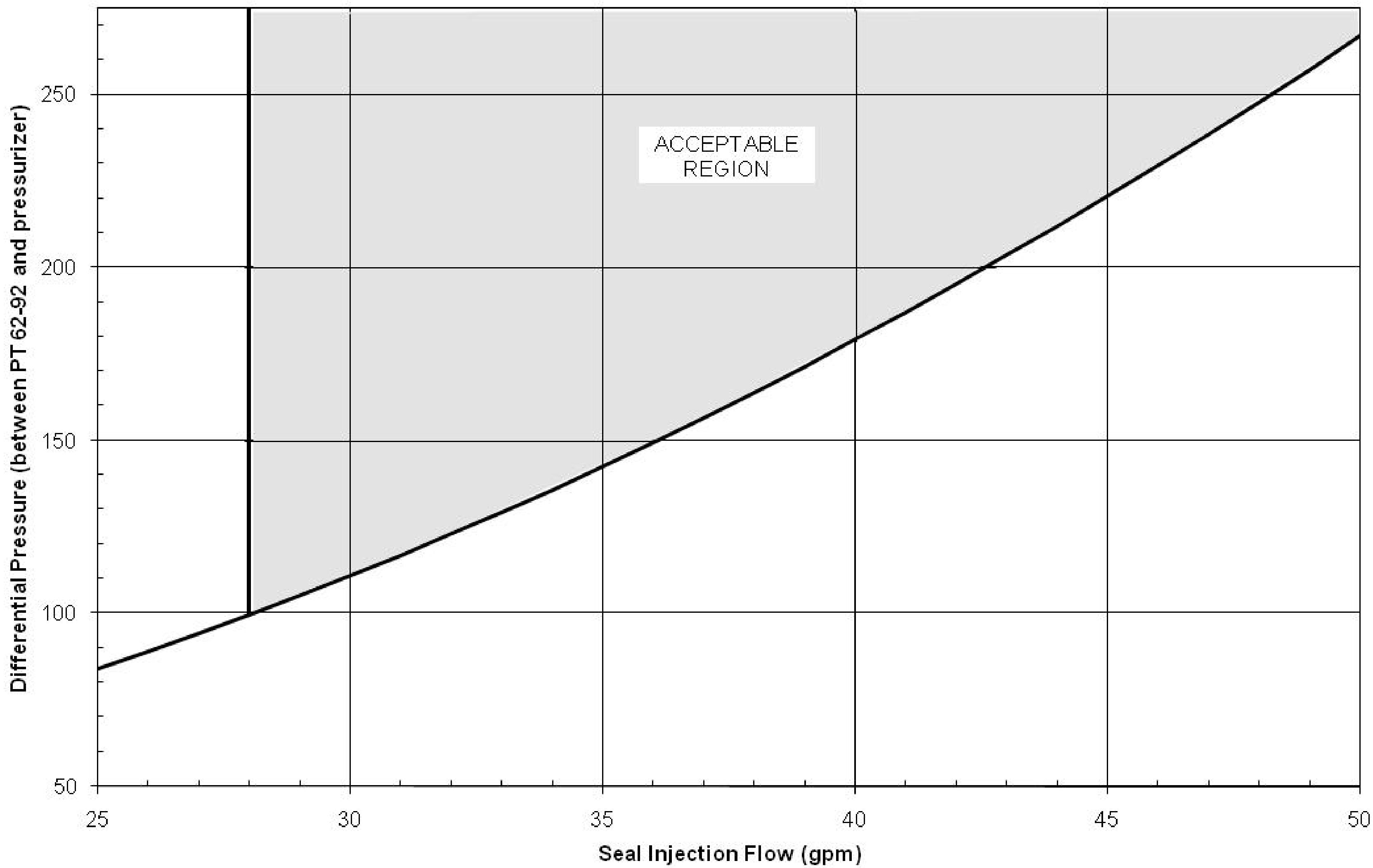
3

Figure 3.5.5-1 (page 1 of 1)
Seal Injection Flow Limits

CTS

3 **INSERT 1**

Figure 3.5.6-1



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3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

3.5.6

LCO 3.5.5 Reactor coolant pump seal injection flow ~~[resistance]~~ shall be ~~[≤ [40] gpm with [centrifugal charging pump discharge header] pressure ≥ [2480] psig and the [charging flow] control valve full open or ≥ [0.2117] ft/gpm^2 or~~ within the limits of Figure 3.5.5-1~~]~~. 1

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Seal injection flow ~~[resistance]~~ not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow ~~[resistance]~~ within limit.	4 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.5.6 * SR 3.5.5.1	-------------------------------NOTE------------------------------ Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at ≥ ~~[~~2215 psig and ≤ 2255 psig~~]~~. ---		1
4.5.6	Verify manual seal injection throttle valves are adjusted to give a flow ~~[resistance] [of ≤ [40 gpm] with [centrifugal charging pump discharge header] pressure ≥ [2480] psig and the [charging flow] control valve full open or ≥ [0.2117] ft/gpm² or~~ within the limit of Figure 3.5.5-1.~~]~~	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

INSERT 1

3

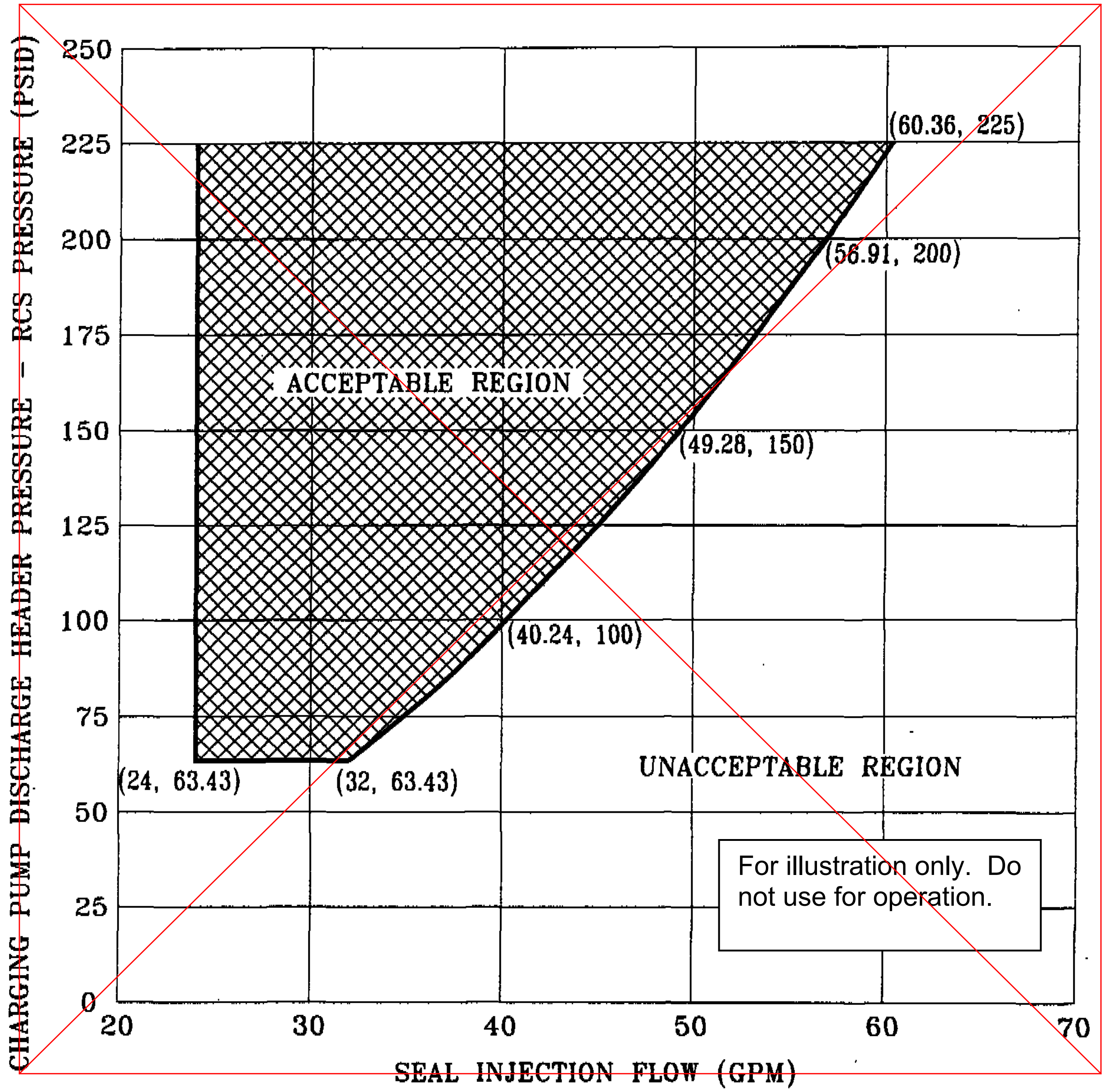
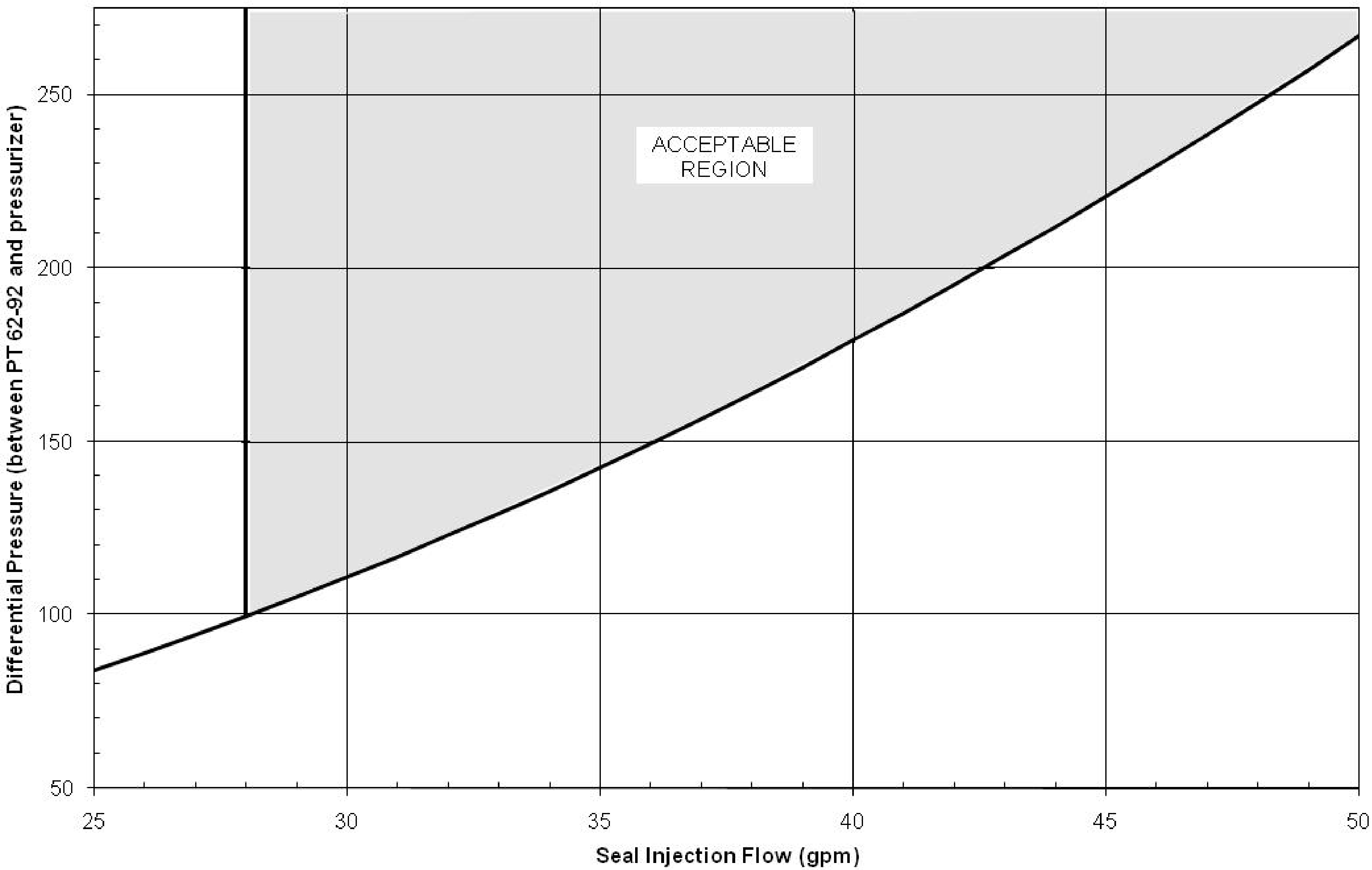


Figure 3.5.5-1 (page 1 of 1)
Seal Injection Flow Limits

3 **INSERT 1**

Figure 3.5.6-1



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JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5, SEAL INJECTION FLOW

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS SR 3.5.5.1 (ITS SR 3.5.5.1) provide two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for this SR under the Surveillance Frequency Control Program.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

~~This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI).~~ The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS). 1

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during ~~SI~~ safety injection (SI). 2

The RCP seal injection flow is restricted by the seal injection line flow ~~[resistance]~~ which is adjusted through positioning of the manual RCP seal injection throttle valves. The RCP seal injection flow ~~[resistance]~~ is determined by measuring the pressurizer pressure, the centrifugal charging pump discharge header pressure, and the RCP seal injection flow rate. 3

The charging flow control valve throttles the centrifugal charging pump discharge header flow as necessary to maintain the programmed level in the pressurizer. The charging flow control valve fails open to ensure that, in the event of either loss of air or loss of control signal to the valve, when the centrifugal charging pumps are supplying charging flow, seal injection flow to the RCP seals is maintained. Positioning of the charging flow control valve may vary during normal plant operating conditions, resulting in a proportional change to RCP seal injection flow. The flow ~~[resistance]~~ provided by RCP seal injection throttle valves will remain fixed when the charging flow control valve is repositioned provided the throttle valve(s) position are not adjusted. 3

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO ensures that seal injection flow ~~[resistance]~~ will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 3

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

~~[The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the value specified in this LCO. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the control valve (charging flow for four loop units and air operated seal injection for three loop units) being full open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.~~

~~OR~~

~~This is accomplished by limiting the seal injection line resistance to a value consistent with the assumptions in the accident analysis. The limit on RCP seal injection flow resistance must be met to assure that the ECCS is OPERABLE. If this limit is not met, the ECCS flow may not be as assumed in the accident analysis. The restriction on seal injection flow is accomplished by maintaining the seal water injection flow resistance ≥ [0.2117] ft/gpm^2. With the seal injection flow resistances within limit, the resulting total seal injection flow will be within the assumptions made for seal flow during accident conditions.~~ 3

BASES

LCO (continued)

~~In order to establish the proper flow line resistance, the centrifugal charging pump discharge header pressure, the RCP seal injection flow rate, and the pressurizer pressure are measured. The line resistance is then determined from those inputs. A reduction in RCP pressure with no concurrent decrease in centrifugal charging pump discharge header pressure would increase the differential pressure across the manual throttle valves, and result in more flow being discharged through the RCP seal injection line. The flow resistance limit assures that when RCS pressure drops during a LOCA and seal injection flow increases in response to the higher differential pressure, the resulting flow will be consistent with the accident analysis.~~ 3

~~OR~~

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection throttle valves (needle valves) to provide a total seal injection flow in the acceptable region of Figure 3.5.5-1 at a given pressure differential between the charging header and the RCS. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Figure 3.5.5-1 ensures that the minimum ECCS flow assumed in the safety analyses is maintained.~~]~~ 2 3

The limit on seal injection flow ~~[resistance]~~ must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses. 3

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow ~~[resistance]~~ limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow ~~[resistance]~~ limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow ~~[resistance]~~ must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance. 3 3 3

BASES

ACTIONS

A.1

With the seal injection flow ~~[resistance]~~ not within its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow ~~[resistance]~~ to within its limit. The operator has 4 hours from the time the flow ~~[resistance]~~ is known to not be within the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow ~~[resistance]~~ within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel. 3 3

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification that the manual seal injection throttle valves are adjusted to give a flow ~~[resistance]~~ within the limit ensures that the ECCS injection flows stay within the safety analysis. A differential pressure is established between the charging header and the RCS, and the total seal injection flow is verified to within the limit determined in accordance with the ECCS safety analysis. ~~[The flow [resistance] shall be verified by confirming seal injection flow ≤ [40] gpm with the RCS at normal operating pressure, the charging flow control valve full open, and the charging header pressure ≥ [2480].~~ 3 3

~~OR~~

The flow ~~[resistance]~~ shall be verified by confirming seal injection flow and differential pressure within the acceptable region of Figure 3.5.5-1. 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The flow resistance shall be ≥ [0.2117] ft/gpm²,]~~ Control valves in the flow path between the charging header and the RCS pressure sensing points must be in their post accident position (e.g., charging flow control valve open) during this Surveillance to correlate with the acceptance criteria. 3

~~[The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.~~

~~OR~~ 4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5 4

at ≥ 2215 psig and ≤ 2255 psig

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized ~~within a ± 20 psig range of normal operating pressure~~. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely. 2

REFERENCES

1. U FSAR, Chapter ~~[~~6~~]~~ and Chapter ~~[~~15~~]~~. 2 3

2. 10 CFR 50.46.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

~~This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI).~~ The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS). 1

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during ~~SI~~. safety injection (SI) 2

The RCP seal injection flow is restricted by the seal injection line flow ~~[resistance]~~ which is adjusted through positioning of the manual RCP seal injection throttle valves. The RCP seal injection flow ~~[resistance]~~ is determined by measuring the pressurizer pressure, the centrifugal charging pump discharge header pressure, and the RCP seal injection flow rate. 3

The charging flow control valve throttles the centrifugal charging pump discharge header flow as necessary to maintain the programmed level in the pressurizer. The charging flow control valve fails open to ensure that, in the event of either loss of air or loss of control signal to the valve, when the centrifugal charging pumps are supplying charging flow, seal injection flow to the RCP seals is maintained. Positioning of the charging flow control valve may vary during normal plant operating conditions, resulting in a proportional change to RCP seal injection flow. The flow ~~[resistance]~~ provided by RCP seal injection throttle valves will remain fixed when the charging flow control valve is repositioned provided the throttle valve(s) position are not adjusted. 3

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

APPLICABLE SAFETY ANALYSES (continued)

This LCO ensures that seal injection flow ~~[resistance]~~ will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 3

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

~~[The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the value specified in this LCO. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the control valve (charging flow for four loop units and air operated seal injection for three loop units) being full open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.~~

~~OR~~

~~This is accomplished by limiting the seal injection line resistance to a value consistent with the assumptions in the accident analysis. The limit on RCP seal injection flow resistance must be met to assure that the ECCS is OPERABLE. If this limit is not met, the ECCS flow may not be as assumed in the accident analysis. The restriction on seal injection flow is accomplished by maintaining the seal water injection flow resistance ≥ [0.2117] ft/gpm^2. With the seal injection flow resistances within limit, the resulting total seal injection flow will be within the assumptions made for seal flow during accident conditions.~~ 3

BASES

LCO (continued)

~~In order to establish the proper flow line resistance, the centrifugal charging pump discharge header pressure, the RCP seal injection flow rate, and the pressurizer pressure are measured. The line resistance is then determined from those inputs. A reduction in RCP pressure with no concurrent decrease in centrifugal charging pump discharge header pressure would increase the differential pressure across the manual throttle valves, and result in more flow being discharged through the RCP seal injection line. The flow resistance limit assures that when RCS pressure drops during a LOCA and seal injection flow increases in response to the higher differential pressure, the resulting flow will be consistent with the accident analysis.~~ 3

~~OR~~

seal injection throttle valves (needle valves) to provide a total

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection flow in the acceptable region of Figure 3.5.5-1 at a given pressure differential between the charging header and the RCS. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Figure 3.5.5-1 ensures that the minimum ECCS flow assumed in the safety analyses is maintained.~~]~~ 2 3

The limit on seal injection flow ~~[resistance]~~ must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses. 3

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow ~~[resistance]~~ limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow ~~[resistance]~~ limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow ~~[resistance]~~ must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance. 3 3 3

BASES

ACTIONS

A.1

With the seal injection flow ~~[resistance]~~ not within its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow ~~[resistance]~~ to within its limit. The operator has 4 hours from the time the flow ~~[resistance]~~ is known to not be within the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow ~~[resistance]~~ within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel. 3 3

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification that the manual seal injection throttle valves are adjusted to give a flow ~~[resistance]~~ within the limit ensures that the ECCS injection flows stay within the safety analysis. A differential pressure is established between the charging header and the RCS, and the total seal injection flow is verified to within the limit determined in accordance with the ECCS safety analysis. ~~[The flow [resistance] shall be verified by confirming seal injection flow ≤ [40] gpm with the RCS at normal operating pressure, the charging flow control valve full open, and the charging header pressure ≥ [2480].~~ 3 3

~~OR~~

The flow ~~[resistance]~~ shall be verified by confirming seal injection flow and differential pressure within the acceptable region of Figure 3.5.5-1. 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

~~The flow resistance shall be ≥ [0.2117] ft/gpm^2.]~~ Control valves in the flow path between the charging header and the RCS pressure sensing points must be in their post accident position (e.g., charging flow control valve open) during this Surveillance to correlate with the acceptance criteria. 3

~~[The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 5 4

at ≥ 2215 psig and ≤ 2255 psig

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized ~~within a ± 20 psig range of normal operating pressure~~. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely. 2

REFERENCES

1. U FSAR, Chapter ~~[~~6~~]~~ and Chapter ~~[~~15~~]~~. 2 3

2. 10 CFR 50.46.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5 BASES, SEAL WATER INJECTION

1. The deleted sentence is not required based on SQN utilizes centrifugal charging pumps for safety injection.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.5.1.1 Bases provide two options for controlling the Frequency of the Surveillance Requirement. SQN is proposing to control the Surveillance Frequency for ITS SR 3.5.1.1 under the Surveillance Frequency Control Program.

5. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.5, SEAL INJECTION FLOW

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

Improved Standard Technical Specifications (ISTS) Not Adopted in the Sequoyah ITS

ISTS 3.5.6, BORON INJECTION TANK (BIT)

~~3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)~~

~~3.5.6 Boron Injection Tank (BIT)~~

~~LCO 3.5.6 The BIT shall be OPERABLE.~~

~~APPLICABILITY: MODES 1, 2, and 3.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. BIT inoperable.~~	~~A.1 Restore BIT to OPERABLE status.~~	~~1 hour~~
~~B. Required Action and associated Completion Time of Condition A not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Borate to SDM specified in COLR.~~	~~6 hours~~
	~~AND~~	
	~~B.3 Restore BIT to OPERABLE status.~~	~~7 days~~
~~C. Required Action and associated Completion Time of Condition B not met.~~	~~C.1 Be in MODE 4.~~	~~12 hours~~

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~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.5.6.1~~	~~Verify BIT borated water temperature is ≥ [145]°F.~~	~~[24 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
~~SR 3.5.6.2~~	~~[Verify BIT borated water volume is ≥ [1100] gallons.~~	~~[7 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
~~SR 3.5.6.3~~	~~Verify BIT boron concentration is ≥ [20,000] ppm and ≤ [22,500] ppm.~~	~~[7 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

1

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

JUSTIFICATION FOR DEVIATIONS
ISTS 3.5.6, BORON INJECTION TANK (BIT)

1. ISTS 3.5.6, Boron Injection Tank (BIT) is not being adopted because Sequoyah Nuclear Plant (SQN) design does not include the BIT. Therefore, ISTS 3.5.6 is not included in the ITS.
   .

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

~~B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)~~

~~B 3.5.6 Boron Injection Tank (BIT)~~

~~BASES~~

~~BACKGROUND~~

~~The BIT is part of the Boron Injection System, which is the primary means of quickly introducing negative reactivity into the Reactor Coolant System (RCS) on a safety injection (SI) signal.~~

~~The main flow path through the Boron Injection System is from the discharge of the centrifugal charging pumps through lines equipped with a flow element and two valves in parallel that open on an SI signal. The valves can be operated from the main control board. The valves and flow elements have main control board indications. Downstream of these valves, the flow enters the BIT (Ref. 1).~~

~~The BIT is a stainless steel tank containing concentrated boric acid. Two trains of strip heaters are mounted on the tank to keep the temperature of the boric acid solution above the precipitation point. The strip heaters are controlled by temperature elements located near the bottom of the BIT. The temperature elements also activate High and Low alarms on the main control board. In addition to the strip heaters on the BIT, there is a recirculation system with a heat tracing system, including the piping section between the motor operated isolation valves, which further ensures that the boric acid stays in solution. The BIT is also equipped with a High Pressure alarm on the main control board. The entire contents of the BIT are injected when required; thus, the contained and deliverable volumes are the same.~~

~~During normal operation, one of the two BIT recirculation pumps takes suction from the boron injection surge tank (BIST) and discharges to the BIT. The solution then returns to the BIST. Normally, one pump is running and one is shut off. On receipt of an SI signal, the running pump shuts off and the air operated valves close. Flow to the BIT is then supplied from the centrifugal charging pumps. The solution of the BIT is injected into the RCS through the RCS cold legs.~~

~~APPLICABLE SAFETY ANALYSES~~

~~During a main steam line break (MSLB) or loss of coolant accident (LOCA), the BIT provides an immediate source of concentrated boric acid that quickly introduces negative reactivity into the RCS.~~

~~The contents of the BIT are not credited for core cooling or immediate boration in the LOCA analysis, but for post LOCA recovery. The BIT maximum boron concentration of [22,500] ppm is used to determine the minimum time for hot leg recirculation switchover. The minimum boron~~

1

~~BASES~~

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~concentration of [20,000] ppm is used to determine the minimum mixed mean sump boron concentration for post LOCA shutdown requirements.~~

~~For the MSLB analysis, the BIT is the primary mechanism for injecting boron into the core to counteract any positive increases in reactivity caused by an RCS cooldown. The analysis uses the minimum boron concentration of the BIT, which also affects both the departure from nucleate boiling and containment design analyses. Reference to the LOCA and MSLB analyses is used to assess changes to the BIT to evaluate their effect on the acceptance limits contained in these analyses.~~

~~The minimum temperature limit of [145]°F for the BIT ensures that the solution does not reach the boric acid precipitation point. The temperature of the solution is monitored and alarmed on the main control board.~~

~~The BIT boron concentration limits are established to ensure that the core remains subcritical during post LOCA recovery. The BIT will counteract any positive increases in reactivity caused by an RCS cooldown.~~

~~The BIT minimum water volume limit of [1100] gallons is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to shut down the core following an MSLB, to determine the hot leg recirculation switchover time, and to safeguard against boron precipitation.~~

~~The BIT satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).~~

~~LCO~~ ~~This LCO establishes the minimum requirements for contained volume, boron concentration, and temperature of the BIT inventory (Ref. 2). This ensures that an adequate supply of borated water is available in the event of a LOCA or MSLB to maintain the reactor subcritical following these accidents.~~

~~To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, and temperature must be met.~~

~~If the equipment used to verify BIT parameters (temperature, volume, and boron concentration) is determined to be inoperable, then the BIT is also inoperable.~~

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~~BASES~~

~~APPLICABILITY~~

~~In MODES 1, 2, and 3, the BIT OPERABILITY requirements are consistent with those of LCO 3.5.2, "ECCS - Operating."~~

~~In MODES 4, 5, and 6, the respective accidents are less severe, so the BIT is not required in these lower MODES.~~

~~ACTIONS~~

~~A.1~~

~~If the required volume is not present in the BIT, both the hot leg recirculation switchover time analysis and the boron precipitation analysis would not be met. Under these conditions, prompt action must be taken to restore the volume to above its required limit to declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.~~

~~The BIT boron concentration is considered in the hot leg recirculation switchover time analysis, the boron precipitation analysis, and the reactivity analysis for an MSLB. If the concentration were not within the required limits, these analyses could not be relied on. Under these conditions, prompt action must be taken to restore the concentration to within its required limits, or the plant must be placed in a MODE in which the BIT is not required.~~

~~The BIT temperature limit is established to ensure that the solution does not reach the boric acid crystallization point. If the temperature of the solution drops below the minimum, prompt action must be taken to raise the temperature and declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.~~

~~The 1 hour Completion Time to restore the BIT to OPERABLE status is consistent with other Completion Times established for loss of a safety function and ensures that the plant will not operate for long periods outside of the safety analyses.~~

~~B.1, B.2, and B.3~~

~~When Required Action A.1 cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions and to be borated to the required SDM without challenging plant systems or operators. Borating to the required SDM assures that the plant is in a safe condition, without need for any additional boration.~~

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~~BASES~~

~~ACTIONS (continued)~~

~~After determining that the BIT is inoperable and the Required Actions of B.1 and B.2 have been completed, the tank must be returned to OPERABLE status within 7 days. These actions ensure that the plant will not be operated with an inoperable BIT for a lengthy period of time. It should be noted, however, that changes to applicable MODES cannot be made until the BIT is restored to OPERABLE status pursuant to the provisions of LCO 3.0.4.~~

~~C.1~~

~~Even though the RCS has been borated to a safe and stable condition as a result of Required Action B.2, either the BIT must be restored to OPERABLE status (Required Action C.1) or the plant must be placed in a condition in which the BIT is not required (MODE 4). The 12 hour Completion Time to reach MODE 4 is reasonable, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.5.6.1~~

~~Verification that the BIT water temperature is at or above the specified minimum temperature will identify a temperature change that would approach the acceptable limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. [The Frequency of 24 hours has been shown to be acceptable through operating experience.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

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~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SR 3.5.6.2~~

~~Verification that the BIT contained volume is above the required limit assures that this volume will be available for quick injection into the RCS. If the volume is too low, the BIT would not provide enough borated water to ensure subcriticality during recirculation or to shut down the core following an MSLB. [Since the BIT volume is normally stable, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

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~~SR 3.5.6.3~~

~~Verification that the boron concentration of the BIT is within the required band ensures that the reactor remains subcritical following a LOCA; it limits return to power following an MSLB, and maintains the resulting sump pH in an acceptable range so that boron precipitation will not occur in the core. In addition, the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.~~

~~The BIT is in a recirculation loop that provides continuous circulation of the boric acid solution through the BIT and the boric acid tank (BAT). There are a number of points along the recirculation loop where local samples can be taken. The actual location used to take a sample of the solution is specified in the plant Surveillance procedures. Sampling from the BAT to verify the concentration of the BIT is not recommended, since this sample may not be homogenous and the boron concentration of the two tanks may differ.~~

~~The sample should be taken from the BIT or from a point in the flow path of the BIT recirculation loop.~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~[The Frequency of 7 days is appropriate and has been shown to be acceptable through operating experience.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

~~REFERENCES~~

~~1. FSAR, Chapter [6] and Chapter [15].~~

~~2. 10 CFR 50.46.~~

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.6 BASES, BORON INJECTION TANK (BIT)

1. ISTS 3.5.6 Bases, "Boron Injection Tank (BIT)" is not included in the Sequoyah Nuclear Plant (SQN) ITS since the Specification, ISTS 3.5.6, has not been included in the SQN ITS.

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ENCLOSURE 2

VOLUME 15

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 4.0
DESIGN FEATURES

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 4.0, DESIGN FEATURES

**Current Technical Specification (CTS) Markup
and Discussion of Changes (DOCs)**

5.0 DESIGN FEATURES

5.1 SITE LOCATION

4.1 The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

~~EXCLUSION AREA~~

~~5.2.1 DELETED~~

~~LOW POPULATION ZONE~~

~~5.1.2 DELETED~~

~~SITE BOUNDARY FOR GASEOUS EFFLUENTS~~

~~5.1.3 DELETED~~

~~SITE BOUNDARY FOR LIQUID EFFLUENTS~~

~~5.1.4 DELETED~~

~~5.2 CONTAINMENT~~

~~CONFIGURATION~~

~~5.2.1 DELETED~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.2.2 DELETED~~

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4.2 5.3 REACTOR CORE

FUEL ASSEMBLIES

4.2.1 5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 1 Operating Cycle 12.

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 DELETED~~

~~VOLUME~~

~~5.4.2 DELETED~~

~~5.5 METEOROLOGICAL TOWER LOCATION~~

~~5.5.1 DELETED~~

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY - SPENT FUEL

4.3.1.1 4.3.1.1.a 5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

4.3.1.1.b a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

4.3.1.1.c b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the equations in Table 5.6-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2. See ITS 3.7.15

Restrictions in Region 1

Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

1) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

2) The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

3) If Region 1 arrays are used in conjunction with Region 2 or Region 3 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

4.3.1.1.b *For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

DESIGN FEATURES

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY - NEW FUEL

4.3.1.2.a 5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 weight percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure 5.6-4. The cells shown
4.3.1.2.d as empty cells in Figure 5.6-4 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain less than or
4.3.1.2.b equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under optimum moderation
4.3.1.2.c conditions.

DRAINAGE

4.3.2 5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

4.3.3 5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

~~5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT~~

~~5.7.1 DELETED~~

Figure 4.3.1.2-1

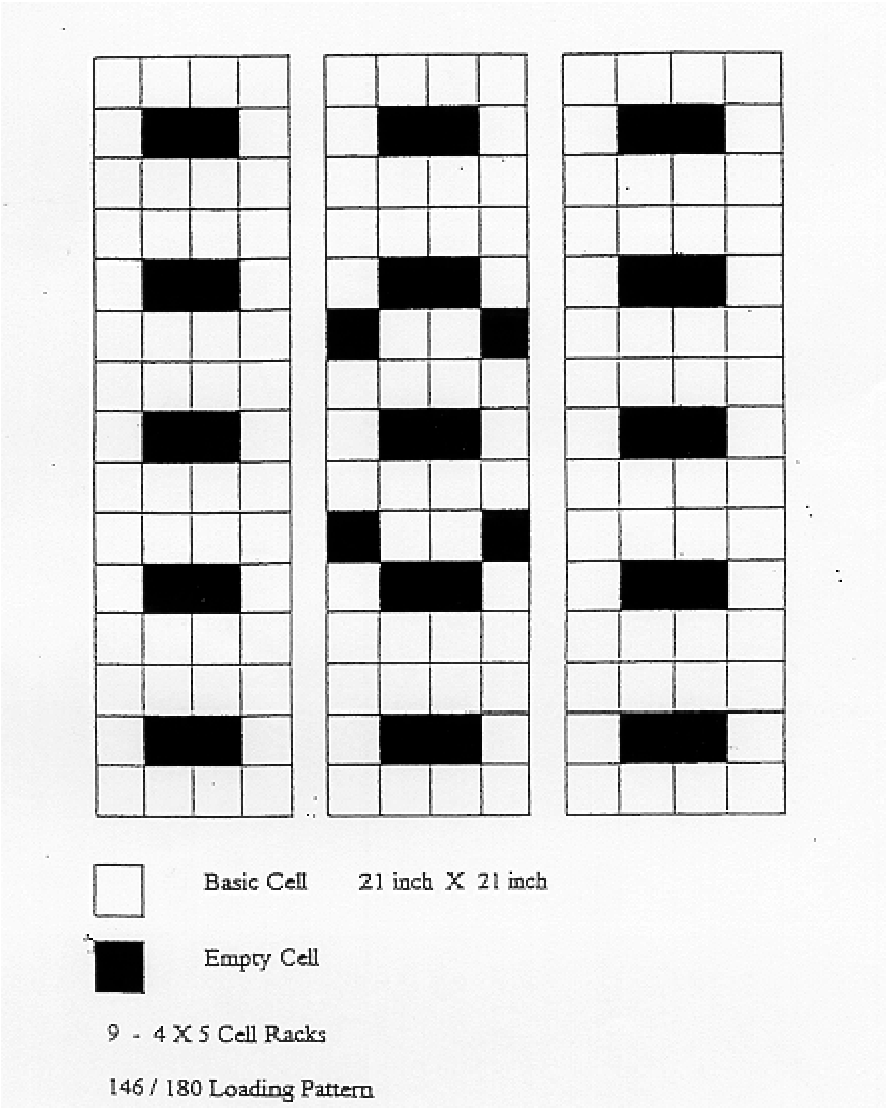

Figure 5.6-4
New Fuel Pit Storage Rack Loading Pattern

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5.0 DESIGN FEATURES

5.1 SITE LOCATION

4.1 The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

~~EXCLUSION AREA~~

~~5.1.1 DELETED~~

~~LOW POPULATION ZONE~~

~~5.1.2 DELETED~~

~~SITE BOUNDARY FOR GASEOUS EFFLUENTS~~

~~5.1.3 DELETED~~

~~SITE BOUNDARY FOR LIQUID EFFLUENTS~~

~~5.1.4 DELETED~~

~~5.2 CONTAINMENT~~

~~5.2.1 DELETED~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.2.2 DELETED~~

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4.2 5.3 REACTOR CORE

FUEL ASSEMBLIES

4.2.1 5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor, as described in the Framatome Cogema Fuels Report BAW-2328, beginning with the Unit 2 Operating Cycle 10 core.

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 DELETED~~

~~VOLUME~~

~~5.4.2 DELETED~~

~~5.5 METEOROLOGICAL TOWER LOCATION~~

~~5.5.1 DELETED~~

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY - SPENT FUEL

4.3.1.1 5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

4.3.1.1.a

4.3.1.1.b a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

4.3.1.1.c b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the equations in Table 5.6-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2. See ITS 3.7.15

Restrictions in Region 1

Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

1) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

2) The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

3) If Region 1 arrays are used in conjunction with Region 2 or Region 3 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

4.3.1.1.b *For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

DESIGN FEATURES

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY - NEW FUEL

4.3.1.2.a 5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 weight percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure 5.6-4. The cells
4.3.1.2.d shown as empty cells in Figure 5.6-4 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain
4.3.1.2.b less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under
4.3.1.2.c optimum moderation conditions.

DRAINAGE

4.3.2 5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

4.3.3 5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

~~5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT~~

~~5.7.1 DELETED~~

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Figure 4.3.1.2-1

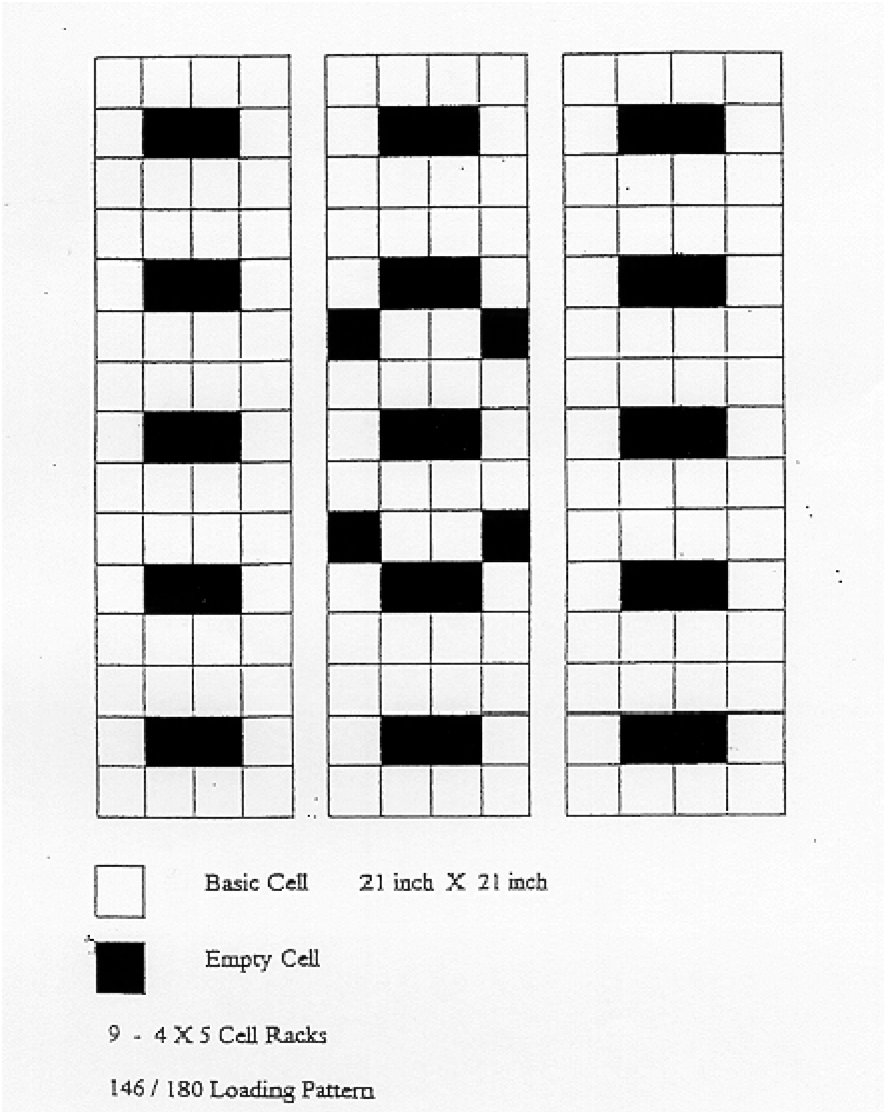

Figure 5.6-4
New Fuel Pit Storage Rack Loading Pattern

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DISCUSSION OF CHANGES
ITS Chapter 4.0, DESIGN FEATURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

4.0 DESIGN FEATURES

5.1 ## 4.1 Site Location

INSERT 1

~~[Text description of site location.]~~ 1

5.3 ## 4.2 Reactor Core

5.3.1 ### 4.2.1 Fuel Assemblies

193 M5 clad Zircaloy

The reactor shall contain ~~[157]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircalloy~~ or ~~ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. INSERT 2 5 1 2

5.3.2 ### 4.2.2 ~~[~~Control Rod~~]~~ Assemblies 1

53 full length and no part length

The reactor core shall contain ~~[48]~~ ~~[~~control rod~~]~~ assemblies. ~~The control material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.~~ INSERT 3 1 2

5.6 ## 4.3 Fuel Storage

5.6 ### 4.3.1 Criticality

5.6.1.1 4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

5.6.1.1 a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent~~,~~ ; 1 3

5.6.1.1.a, Footnote * b. ~~k_{eff} ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ INSERT 4 2

5.6.1.1.b ~~[~~ c. A nominal ~~[9.15]~~ 8.972 inch center to center distance between fuel assemblies placed in ~~[~~the high density fuel storage racks~~],]~~ . 1

~~[d. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks],]~~ 1

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1 **INSERT 1**

5.1 The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

2 **INSERT 2**

5.3.1 Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 1 Operating Cycle 12.

2 **INSERT 3**

5.3.2 The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

2 **INSERT 4**

5.6.1.1.a, Footnote * A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron. For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident; and

4.0 DESIGN FEATURES

5.6 4.3 Fuel Storage (continued)

~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s), and]~~ 1

~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~ 1

5.6.1.2 4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

5.6.1.2 a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent~~,~~ ; 1 3

5.6.1.2 b. $k_{eff} \leq 0.95$ if fully flooded with unborated water~~, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ ; 3 2 1

5.6.1.2 c. $k_{eff} \leq 0.98$ ~~if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ under optimum moderation conditions; and 2 1 3

5.6.1.2 d. ~~A nominal [10.95] inch center to center distance between fuel assemblies placed in the storage racks.~~ INSERT 5 1 2

5.6. 2 4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23~~ 722 ft~~]~~. 1

5.6.3 4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1737]~~ 2091 fuel assemblies. INSERT 6 1 2

INSERT 7 2

2 **INSERT 5**

5.6.1.2 The arrangement of 146 storage locations shown in Figure 4.3.1.2-1. The cells shown as empty cells in Figure 4.3.1.2-1 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur.

2 **INSERT 6**

5.6.3 In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

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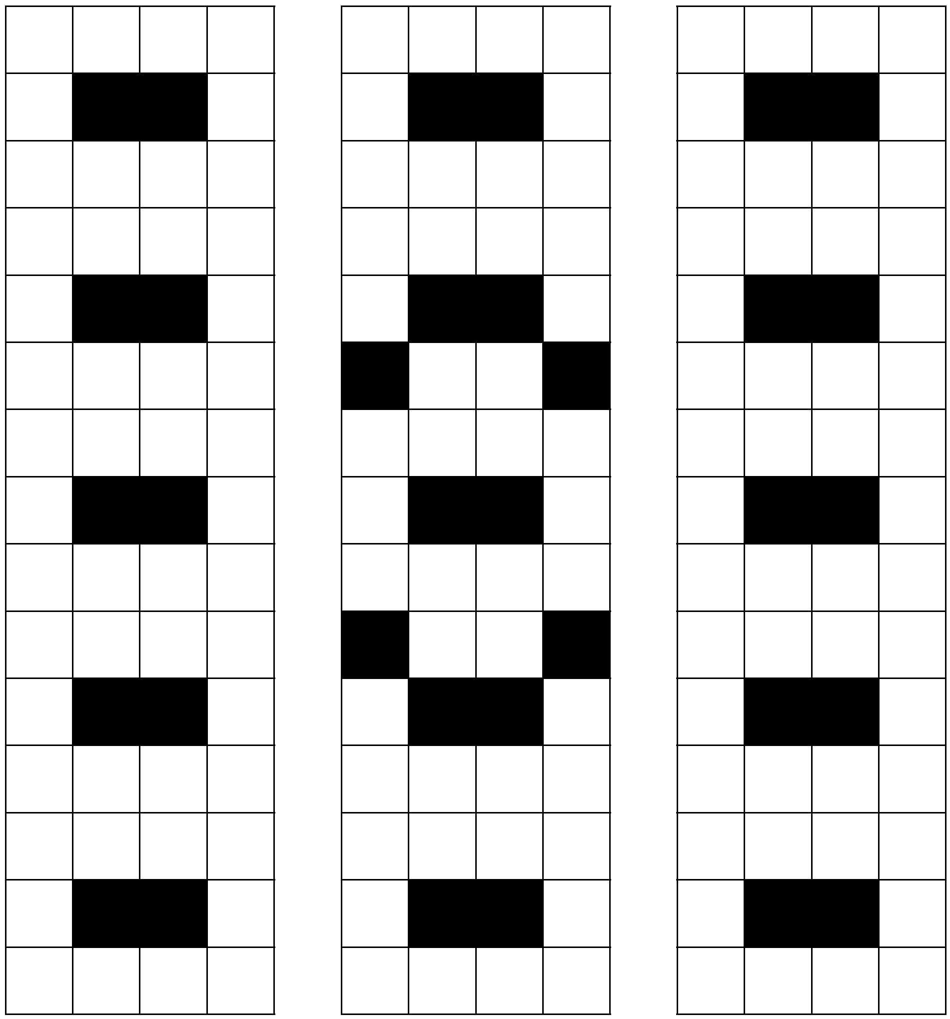
2 **INSERT 7**



Basic Cell 21 inch X 21 inch

Empty Cell

9 – 4 X 5 Cell Racks

146 / 180 Loading Pattern

Figure 5.6-4

Figure 4.3.1.2-1
New Fuel Storage Rack Loading Pattern

4.0 DESIGN FEATURES

5.1 ## 4.1 Site Location

INSERT 1

~~[Text description of site location.]~~ 1

5.3 ## 4.2 Reactor Core

5.3.1 ### 4.2.1 Fuel Assemblies

193

M5 clad

Zircaloy

The reactor shall contain ~~[157]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircalloy~~ or ~~ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. INSERT 2 5 1 2

5.3.2 ### 4.2.2 ~~[~~Control Rod~~]~~ Assemblies 1

53 full length and no part length

The reactor core shall contain ~~[48]~~ ~~[~~control rod~~]~~ assemblies. ~~The control material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.~~ INSERT 3 1 2

5.6 ## 4.3 Fuel Storage

5.6 ### 4.3.1 Criticality

5.6.1.1 4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

5.6.1.1 a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent~~,~~ ; 1 3

5.6.1.1.a, Footnote * b. ~~k_{eff} ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ INSERT 4 2

5.6.1.1.b ~~[~~ c. A nominal ~~[9.15]~~ 8.972 inch center to center distance between fuel assemblies placed in ~~[~~the high density fuel storage racks~~],]~~ . 1

~~[d. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks],]~~ 1

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CTS 4.0

1 **INSERT 1**

5.1 The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

2 **INSERT 2**

5.3.1 Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the ~~Unit 1 Operating Cycle 12~~ Unit 2 Operating Cycle 10 core.

SII

2 **INSERT 3**

5.3.2 The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

2 **INSERT 4**

5.6.1.1.a, Footnote * A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron. For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident; and

4.0 DESIGN FEATURES

5.6 4.3 Fuel Storage (continued)

~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s), and]~~ 1

~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~ 1

5.6.1.2 4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

5.6.1.2 a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent~~,~~ ; 1 3

5.6.1.2 b. $k_{eff} \leq 0.95$ if fully flooded with unborated water~~, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ ; 2 3 1

5.6.1.2 c. $k_{eff} \leq 0.98$ ~~if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],~~ under optimum moderation conditions; and 2 1 3

5.6.1.2 d. ~~A nominal [10.95] inch center to center distance between fuel assemblies placed in the storage racks.~~ INSERT 5 1 2

5.6. 2 4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23~~ 722 ft~~]~~. 1

5.6.3 4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1737]~~ 2091 fuel assemblies. INSERT 6 1 2

INSERT 7 2

2 **INSERT 5**

5.6.1.2 The arrangement of 146 storage locations shown in Figure 4.3.1.2-1. The cells shown as empty cells in Figure 4.3.1.2-1 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur.

2 **INSERT 6**

5.6.3 In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

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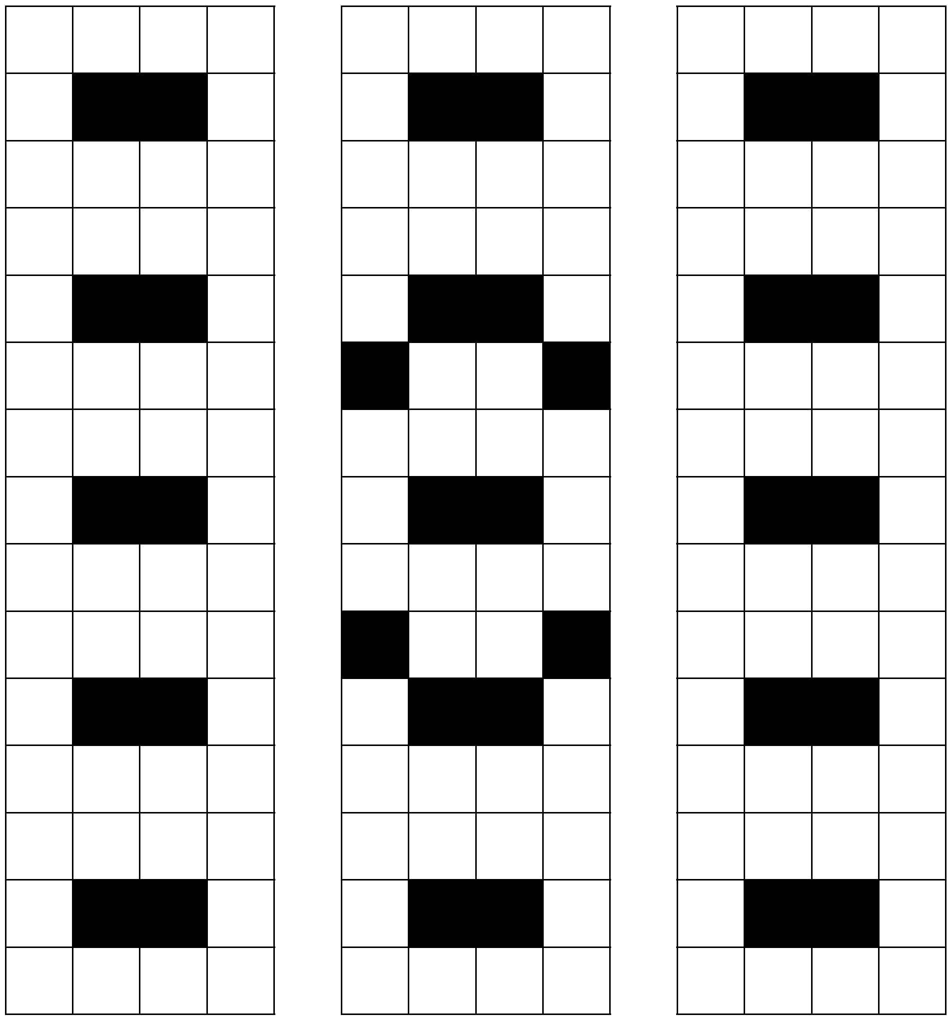
2 **INSERT 7**



Basic Cell 21 inch X 21 inch

Empty Cell

9 – 4 X 5 Cell Racks

146 / 180 Loading Pattern

Figure 5.6-4

Figure 4.3.1.2-1
New Fuel Storage Rack Loading Pattern

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JUSTIFICATION FOR DEVIATIONS
ITS Chapter 4.0, DESIGN FEATURES

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS 4.0 has been changed to address Sequoyah Nuclear Plant (SQN) site specific requirements for fuel assemblies, control rod assemblies, and fuel storage. This change is acceptable because it reflects the current licensing basis.

3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. Typographical/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS Chapter 4.0, DESIGN FEATURES

There are no specific No Significant Hazards Considerations for this Specification.

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ENCLOSURE 2

VOLUME 16

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0
ADMINISTRATIVE CONTROLS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 5.1, RESPONSIBILITY

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

ITS

A01

ITS 5.1

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

LA01

5.1.1 6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

INSERT 1 M01

5.1.2 6.1.2 The Shift Manager (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function.

M02

LA01

LA02

~~6.1.3 The Chief Nuclear Officer is responsible for the safe operation of all TVA Nuclear Power Plants~~.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

b. The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

See ITS 5.2

c. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 FACILITY STAFF

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each unit for which a reactor is operating in MODES 1, 2, 3, or 4. With both units shutdown or defueled, a total of three non-licensed operators are required for the two units.

b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Sections 6.2.2.a and 6.2.2.h for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

M01 INSERT 1

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

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ITS A01 ITS 5.1

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

5.1.1 6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. LA01

INSERT 1 M01

5.1.2 6.1.2 The Shift Manager (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function. LA01 M02

~~6.1.3 The Chief Nuclear Officer is responsible for the safe operation of all TVA Nuclear Power Plants.~~ LA02

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

b. The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant Manager shall be responsible for overall unit safe operation, and shall have control over those onsite resources necessary for safe operation and maintenance of the plant. See ITS 5.2

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 FACILITY STAFF

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each unit for which a reactor is operating in MODES 1, 2, 3, or 4. With both units shutdown or defueled, a total of three non-licensed operators are required for the two units.

b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Sections 6.2.2.a and 6.2.2.h for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

M01 INSERT 1

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

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DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITIES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 6.1.1 states that the Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. ITS 5.1.1 states that the plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. Additionally, it requires that the plant manager or his designee approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety. This changes the CTS by adding an approval requirement for the plant manager or his designee.

The purpose of the ITS 5.1.1 requirement is to provide additional assurance that the plant manager has direct responsibility for overall unit operation. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with CTS 6.2.1.c (ITS 5.2.1.b) requirement that the plant manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant. This change is designated as more restrictive because it adds a requirement for the plant manager or his designee to the CTS.

M02 CTS 6.1.2 allows a designated individual to assume the responsibility for the control room command function when the Shift Manager is absent from the Control Room. ITS 5.1.2 provides the allowance for the designated individual to assume the responsibility for the control room command function, but provides additional requirements for the designated individual. In MODE 1, 2, 3, or 4, ITS 5.1.2 requires the designated individual to hold an active Senior Reactor Operator license. In MODE 5 or 6, ITS 5.1.2 requires the designated individual to hold an active Senior Reactor Operator license or Reactor Operator license. This changes the CTS by adding qualification requirements for the designated individual that assumes the control room command function.

The purpose of the ITS 5.1.2 requirement is to ensure that the control room command function is maintained. This change is acceptable because the additional requirements ensure that the designated individual assuming the control room command functions meets the appropriate qualification

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITIES

requirements. This change is designated as more restrictive because it adds qualification requirements for the designated individual that assumes the control room command function to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.1.1 uses the title "Plant Manager" and CTS 6.1.2 uses the title "Shift Manager." ITS 5.1.1 uses the generic title "plant manager" and ITS 5.1.2 uses the generic title "shift manager." This changes the CTS by moving the specific organizational titles to the Nuclear Power Organization Topical Report (TVA-NPOD89-A) and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific SQN organizational titles out of the Technical Specifications is consistent with the NRC letter from C Grimes to the Owners Group Technical Specification Committee Chairman, dated November 10, 1994. The various requirements of the plant manager and shift manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the Nuclear Power Organization Topical Report (TVA-NPOD89-A) as described in ITS 5.2.1.a. Any changes to the Nuclear Power Organization Topical Report (TVA-NPOD89-A) are made under 10 CFR 50.54(a)(3), which ensures that changes are properly evaluated. This change is a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.1.3 states that the Chief Nuclear Officer is responsible for the safe operation of all TVA Nuclear Power Plants. ITS 5.1 does not contain this requirement. This changes the CTS by moving the requirements of the Chief Nuclear Officer to the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

The removal of these details, which are related to meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because the removed information will be adequately controlled in the UFSAR. Changes to the Nuclear Power Organization Topical Report (TVA-NPOD89-A) are made under 10 CFR 50.54(a)(3), which ensures

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DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITIES

that changes are properly evaluated. This change is a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

~~REVIEWER'S NOTES~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~ 1

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M01 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

6.1.2 5.1.2 The ~~[Shift Supervisor (SS)]~~ shift manager shall be responsible for the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function. 2 2

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

~~REVIEWER'S NOTES~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~ 1

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M01 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

6.1.2 5.1.2 The ~~[Shift Supervisor (SS)]~~ shift manager shall be responsible for the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function. 2 2

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 2

ITS 5.2, ORGANIZATION

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

ITS A01 ITS 5.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function.

6.1.3 The Chief Nuclear Officer is responsible for the safe operation of all TVA Nuclear Power Plants.

See ITS 5.1

6.2 ORGANIZATION

5.2.1 6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

, respectively A01

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

and established throughout A01

5.2.1.a a. Lines of authority, responsibility, and communication shall be ~~established and~~ defined ~~from the~~ highest management levels, ~~through~~ intermediate levels ~~to~~ and ~~including~~ all operating organization positions. These relationships shall be documented and updated, as appropriate, in ~~the form of~~ organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

, A01

, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, M01

A specified corporate officer LA01

5.2.1.c b. ~~The Chief Nuclear Officer~~ shall have corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

LA01

of the plant A01

5.2.1.b c. The Plant Manager shall be responsible for overall ~~unit~~ safe operation and shall have control over those onsite ~~resources~~ necessary for safe operation and maintenance of the plant.

activities

A01

or perform A01

5.2.1.d d. The individuals who train the operating staff ~~and those who~~ carry out health physics ~~and~~ quality assurance functions may report to the appropriate onsite manager; however, ~~they~~ shall have sufficient organizational freedom to ensure their independence from operating pressures.

these individuals

Unit A01

5.2.2 6.2.2 ~~FACILITY~~ STAFF

The unit staff organization shall include the following:

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each unit for which a reactor is operating in MODES 1, 2, 3, or 4. With both units shutdown or defueled, a total of three non-licensed operators are required for the two units.

5.2.2.b b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Sections 6.2.2.a and 6.2.2.h for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

A01

ITS ITS 5.2

ADMINISTRATIVE CONTROLS

5.2.2,c c. A Radiological Control technician# shall be onsite when fuel is in the reactor.

~~d. DELETED~~

~~e. DELETED~~

5.2.2.d f. The Operations Superintendent shall hold a Senior Reactor Operator license.

~~g. DELETED~~

5.2.2.e h. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

A01

position vacant not more than

5.2.2.c #The ~~Radiological Control technician~~ may be ~~offsite~~ for ~~a period of time not to exceed~~ 2 hours in order to ~~accommodate~~ unexpected absence provided immediate action is taken to fill the required positions.

provide for

A01

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ITS

A01

ITS 5.2

~~Table 6.2-1~~
~~MINIMUM SHIFT CREW COMPOSITION~~
~~WITH UNIT 2 IN MODE 5 OR 6 OR DE-FUELED~~

~~THIS PAGE INTENTIONALLY DELETED~~

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~~TABLE 6.2-1 (Continued)~~

~~TABLE NOTATION~~

~~THIS PAGE INTENTIONALLY DELETED~~

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~~6.2.3 INDEPENDENT SAFETY ENGINEERING (ISE) (DELETED)~~

~~6.2.4 SHIFT TECHNICAL ADVISOR (STA) (DELETED)~~

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions in Regulatory Guide 1.8, Revision 2 (April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

See ITS 5.3

6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of TS 6.3.1, perform the functions described in 10 CFR 50.54(m).

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A01

ITS | ITS 5.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function.

6.1.3 The Chief Nuclear Officer is responsible for the safe operation of all TVA Nuclear Power Plants.

See ITS 5.1

6.2 ORGANIZATION

5.2.1 6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

An onsite and an offsite organization shall be established for unit operation and corporate management, respectively (A01). The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be ~~established and~~ defined and established throughout ~~from the~~ highest management levels, ~~through~~ intermediate levels ~~to~~ and ~~including~~ all operating organization positions. These relationships shall be documented and updated, as appropriate, ~~in the form~~ of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements , including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, (M01) shall be documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A). (A01)

5.2.1.c b. ~~The Chief Nuclear Officer~~ A specified corporate officer (LA01) shall have corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured. (LA01)

5.2.1.b c. The Plant Manager shall be responsible for overall ~~unit~~ safe operation of the plant (A01) and shall have control over those onsite ~~resources~~ activities necessary for safe operation and maintenance of the plant.

5.2.1.d d. The individuals who train the operating staff ~~and those who~~ or perform (A01) carry out health physics ~~and~~ quality assurance functions may report to the appropriate onsite manager; however, ~~they~~ these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 6.2.2 ~~FACILITY~~ Unit STAFF (A01)

The unit staff organization shall include the following: (A01)

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each unit for which a reactor is operating in MODES 1, 2, 3, or 4. With both units shutdown or defueled, a total of three non-licensed operators are required for the two units.

5.2.2.b b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Sections 6.2.2.a and 6.2.2.h for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

ADMINISTRATIVE CONTROLS

5.2.2,c c. A Radiological Control technician# shall be onsite when fuel is in the reactor.

~~d. DELETED~~

~~e. DELETED~~

5.2.2.d f. The Operations Superintendent shall hold a Senior Reactor Operator license.

~~g. DELETED~~

5.2.2.e h. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

A01

position vacant not more than

5.2.2.c # The ~~Radiological Control technician~~ position may be ~~offsite~~ vacant for ~~a period of time not to exceed~~ not more than 2 hours in ~~01~~order to ~~accommodate~~ provide for unexpected absence provided immediate action is taken to fill the required positions.

A01

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~~TABLE 6.2-1~~
~~MINIMUM SHIFT CREW COMPOSITION~~
~~WITH UNIT 1 IN MODE 5 OR 6 OR DE-FUELED~~

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~~TABLE 6.2-1 (Continued)~~

~~TABLE NOTATION~~

~~THIS PAGE INTENTIONALLY DELETED~~

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A01

ITS

ITS 5.2

~~6.2.3 INDEPENDENT SAFETY ENGINEERING (ISE) (DELETED)~~

~~6.2.4 SHIFT TECHNICAL ADVISOR (STA) (DELETED)~~

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions in Regulatory Guide 1.8, Revision 2 (April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

See ITS 5.3

6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of TS 6.3.1, perform the functions described in 10 CFR 50.54(m).

~~6.4 TRAINING~~

~~6.4.1 DELETED~~

~~6.5 REVIEW AND AUDIT~~

~~6.5.0 DELETED~~

~~6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)~~

~~6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)~~

~~6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)~~

~~6.5.3 RADIOLOGICAL ASSESSMENT REVIEW COMMITTEE (RARC) (DELETED~~

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

M01 CTS 6.2.1.a regarding documentation and updating of the relationships between operating organization position, requires the organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions to be documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A). ITS 5.2.1.a states "These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Nuclear Power Organization Topical Report (TVA NPOD89-A). This changes the CTS by requiring that the specific SQN organizational titles be specified in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

This change is acceptable because specifying the relationship of the specific SQN organizational titles to the generic titles used in the Technical Specifications and organizational positions, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions used in the Technical Specifications and industry standards in the Nuclear Power Organization Topical Report (TVA-NPOD89-A) continues to ensure that organizational positions and associated responsibilities will be maintained. This change adds the requirements to the Technical Specifications. This change is designated as more restrictive because it requires additional information be maintained in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.2.1.b uses the title "Chief Nuclear Officer," and CTS 6.2.1.c uses the title "Plant Manager." ITS 5.2.1.b uses the generic title

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

"plant manager," and ITS 5.2.1.c uses the generic title "A specified corporate officer." This changes the CTS by moving the specific SQN organizational titles to the Nuclear Power Organization Topical Report (TVA-NPOD89-A) and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements from the Technical Specifications, is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific SQN organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairman, dated November 10, 1994. The various requirements of the plant manager and the specified corporate officer are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the Nuclear Power Organization Topical Report (TVA-NPOD89-A). Any changes to the Nuclear Power Organization Topical Report (TVA-NPOD89-A) will be made under 10 CFR 50.54(a)(3) which will ensure the changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to meeting Technical Specification requirements is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan],~~ Nuclear Power Organization Topical Report (TVA-NPOD89-A); 1 2

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~,~~ ; 2

6.2.1.b c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~,~~ ; and 2

6.2.1.d d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel unit and an additional non-licensed operator shall be assigned for each ~~control room from which a reactor is~~ operating in MODE~~S~~ 1, 2, 3, or 4. 4 2

~~-----REVIEWER'S NOTE-----~~
~~Two unit sites~~ With both units shutdown or defueled , ~~require~~ a total of three non-licensed operators are required for the two units. 2 3
~~----------~~

5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 5

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.f d. The operations ~~manager or assistant operations manager~~ superintendent shall hold an SRO license. 4

6.2.2.h e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

CTS

Organization
5.2

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan],~~ Nuclear Power Organization Topical Report (TVA-NPOD89-A); 1 2

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~,~~; 2

6.2.1.b c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~,~~; and 2

6.2.1.d d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel unit and an additional non-licensed operator shall be assigned for each ~~control room from which a reactor is~~ operating in MODE~~S~~ 1, 2, 3, or 4. 4 2

~~-----REVIEWER'S NOTE-----~~
~~Two unit sites~~ With both units shutdown or defueled, ~~require~~ are required a total of three non-licensed operators for the two units.
~~----------~~ 3 2

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5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 5

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.f d. The operations superintendent ~~manager or assistant operations manager~~ shall hold an SRO license. 4

6.2.2.h e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. ISTS 5.2.1.a contains a Reviewer's Note that allows two units with both units shutdown or defueled to have a total of three non-licensed operators for the two units. This Note applies to Sequoyah Nuclear Plant (SQN) since it is a two unit plant. Additionally, CTS 6.2.2.a contains this same statement. Therefore, the Reviewer's Note has been deleted and the information contained in the note has been added to ITS 5.2.1.a.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. Grammatical/editorial change made for enhanced clarity.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 5.3, UNIT STAFF QUALIFICATIONS

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

A01

ITS ITS 5.3

6.2.3 INDEPENDENT SAFETY ENGINEERING (ISE) (DELETED)

6.2.4 SHIFT TECHNICAL ADVISOR (STA) (DELETED)

See ITS 5.2

5.3 6.3 FACILITY STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions in Regulatory Guide 1.8, Revision 2 (April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of TS 6.3.1, perform the functions described in 10 CFR 50.54(m).

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A01

ITS ITS 5.3

~~6.4 TRAINING~~

~~6.4.1 DELETED~~

~~6.5 REVIEW AND AUDIT~~

~~6.5.0 DELETED~~

~~6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)~~

~~6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)~~

~~6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)~~

~~6.5.3 THIS SPECIFICATION IS DELETED~~

~~6.6 REPORTABLE EVENT ACTION (DELETED)~~

~~6.7 SAFETY LIMIT VIOLATION (DELETED)~~

6.8 PROCEDURES & PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

See ITS 5.4

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6.2.3 INDEPENDENT SAFETY ENGINEERING (ISE) (DELETED)

6.2.4 SHIFT TECHNICAL ADVISOR (STA) (DELETED)

See ITS 5.2

5.3 6.3 FACILITY STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions in Regulatory Guide 1.8, Revision 2 (April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of TS 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 TRAINING

6.4.1 DELETED

6.5 REVIEW AND AUDIT

6.5.0 DELETED

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)

6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)

6.5.3 RADIOLOGICAL ASSESSMENT REVIEW COMMITTEE (RARC) (DELETED

See ITS 5.2

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6.6 REPORTABLE EVENT ACTION (DELETED)

6.7 SAFETY LIMIT VIOLATION (DELETED)

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. DELETED

e. DELETED

f. Fire Protection Program implementation.

g. DELETED

See ITS 5.4

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DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

6.3 5.3 Unit Staff Qualifications

~~-----REVIEWER'S NOTE-----~~
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~
~~-----~~

1

6.3.1 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[~~Regulatory Guide 1.8, Revision 2, ~~1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ INSERT 1

2

6.3.2 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

2 **INSERT 1**

(April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

5.0 ADMINISTRATIVE CONTROLS

6.3 ## 5.3 Unit Staff Qualifications

~~REVIEWER'S NOTE~~

~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ 1

6.3.1 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[~~Regulatory Guide 1.8, Revision 2, ~~1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ INSERT 1 2

6.3.2 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

2 **INSERT 1**

(April 1987) for all new personnel qualifying on positions identified in Regulatory Position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (May 1977).

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JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 4

ITS 5.4, PROCEDURES

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

A01

ITS ITS 5.4

6.4 TRAINING

6.4.1 DELETED

6.5 REVIEW AND AUDIT

6.5.0 DELETED

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)

6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)

6.5.3 THIS SPECIFICATION IS DELETED

6.6 REPORTABLE EVENT ACTION (DELETED)

6.7 SAFETY LIMIT VIOLATION (DELETED)

See ITS 5.3

5.4 6.8 PROCEDURES & PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

Add proposed Specification 5.4.1.b M01

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ADMINISTRATIVE CONTROLS

~~b. Refueling operations.~~ A02

~~c. Surveillance and test activities of safety-related equipment.~~ A02

~~d. DELETED~~

~~e. DELETED~~

5.4.1.d f. Fire Protection Program implementation.

~~g. DELETED~~

5.4.1.c h. Quality Assurance Program for effluent and environmental monitoring~~, using the guidance contained in Regulatory Guide 4.15, December 1977, or Regulatory Guide 1.21, Rev. 1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975~~. LA01

~~i. OFFSITE DOSE CALCULATION MANUAL implementation.~~ A03

~~6.8.2 DELETED~~ Add proposed Specification 5.4.1.e M02

~~6.8.3 DELETED~~

6.8.4 The following programs shall be established, implemented, and maintained.

a. Primary Coolant Sources Outside Containment

A program to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The

See ITS 5.5

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ITS A01 ITS 5.4

ADMINISTRATIVE CONTROLS

6.6 REPORTABLE EVENT ACTION (DELETED)

6.7 SAFETY LIMIT VIOLATION (DELETED)

See ITS 5.3

5.4 6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

Add proposed Specification 5.4.1.b M01

~~b. Refueling operations.~~ A02

~~c. Surveillance and test activities of safety related equipment.~~ A02

~~d. DELETED~~

~~e. DELETED~~

5.4.1.d f. Fire Protection Program implementation.

~~g. DELETED~~

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A01

ITS ITS 5.4

5.4.1.c h. Quality Assurance Program for effluent and environmental monitoring~~, using the guidance contained in Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21, Rev. 1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975~~. LA01

~~i. OFFSITE DOSE CALCULATION MANUAL implementation.~~ A03

Add proposed Specification 5.4.1.e M02

~~6.8.2 DELETED~~

~~6.8.3 DELETED~~

6.8.4 The following programs shall be established, implemented, and maintained.

a. Primary Coolant Sources Outside Containment

A program to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at lease once per 18 months.

The provisions of SR 4.0.2 are applicable

b. In-Plant Radiation Monitoring (DELETED)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

See ITS 5.5

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off-control point chemistry conditions,

(vi) Procedures identifying (a) the authority responsible for the interpretation of the data; and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Deleted

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 6.8.1.b requires written procedures be established, implemented and maintained covering refueling operations. CTS 6.8.1.c requires written procedures be established, implemented and maintained covering surveillance and test activities of safety-related equipment. ITS 5.4.1 requires written procedures shall be established, implemented, and maintained to the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. This changes the CTS by removing the specific wording of CTS 6.8.1.b and CTS 6.8.1.c.

This change is acceptable because the recommendations of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 already require procedures for refueling operations and surveillance tests for safety related activities. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 6.8.1.i requires written procedures be established, implemented and maintained for the OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation. ITS 5.4.1 requires procedures for various activities, but does not specifically list the ODCM. This changes the CTS by removing the specific requirement for written procedures to implement the ODCM.

This change is acceptable because implementing procedures for the ODCM are required by ITS 5.4.1.e. ITS 5.4.1.e (as described in DOC M02) requires that written procedures be established, implemented and maintained for all programs and manuals listed in ITS 5.5. ITS 5.5 includes the ODCM. Therefore, it is not necessary to specifically identify each program in ITS 5.4.1. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS 5.4.1.b requires that written procedures shall be established, implemented, and maintained for the emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. The CTS does not include this requirement. This changes the CTS by adopting a new requirement for emergency operating procedures.

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DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

The purpose of ITS 5.4.1.b is to ensure that written procedures are established, implemented, and maintained covering the emergency operating procedures to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This change is acceptable because it is consistent with an existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, for emergency operating procedures. This change is designated as more restrictive because it imposes a new requirement for procedures within the Technical Specifications.

M02 ITS 5.4.1.e requires that written procedures shall be established, implemented, and maintained for all programs specified in Specification 5.5. The CTS does not include this requirement for any program except the OFFSITE DOSE CALCULATION MANUAL. This changes the CTS by adopting a new requirement for procedures to address all programs described in ITS 5.5.

The purpose of ITS 5.4.1.e is to ensure that written procedures are established, implemented, and maintained covering all programs specified in ITS 5.5. This change is acceptable because it requires written procedures, including proper procedure control to address programs required by ITS 5.5. This change is designated as more restrictive because it imposes new requirements for procedures within the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 6.8.1.h requires written procedures be established, implemented and maintained covering the Quality Assurance Program for effluent and environmental monitoring, "using the guidance in Regulatory Guide 4.15, December 1977, or Regulatory Guide 1.21, Revision 1, 1974, and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the Regulatory Guides to the Nuclear Quality Assurance Program (NQAP).

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for written procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these types of procedural details will be adequately controlled in the NQAP. Any changes to the NQAP are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

6.8

5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978~~,~~ ; 1

DOC M01 b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~],~~ ; 2 3 1 4

program

6.8.1.h c. Quality assurance for effluent and environmental monitoring~~,~~ ; 1

6.8.1.f d. Fire Protection Program implementation~~,~~ ; and 1

6.8.1.i DOC M02 e. All programs specified in Specification 5.5.

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5.0 ADMINISTRATIVE CONTROLS

6.8 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978~~,~~ ; 1

DOC M01 b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~],~~ ; 2 3 1 4

6.8.1.h c. Quality assurance program for effluent and environmental monitoring~~,~~ ; 1

6.8.1.f d. Fire Protection Program implementation~~,~~ ; and 1

6.8.1.i DOC M02 e. All programs specified in Specification 5.5.

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

2. Typographical/grammatical error corrected.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 5.5, PROGRAMS AND MANUALS

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

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

c. Reactor coolant system leakage through a steam generator to the secondary system (primary to secondary leakage). See ITS 1.0

MEMBER(S) OF THE PUBLIC

1.17 DELETED

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

5.5.1.a

5.5.1.b

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1. See ITS 1.0

PHYSICS TESTS

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

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ITS A01 ITS 5.5

SURVEILLANCE REQUIREMENTS (Continued)

4.0.3 (Continued)

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be entered.

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. See ITS 3.0

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

5.5.6 4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1, 2 and 3 ~~components~~ pumps and valves shall be as follows: LA01

~~Inservice Inspection Program~~

~~This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:~~

~~a.~~ ~~Provisions that inservice testing of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;~~ LA01

~~b.~~ ~~The provisions of SR 4.0.2 are applicable to the frequencies for performing inservice inspection activities;~~

5.5.5 c. Inspection of each reactor coolant pump flywheel per the recommendation of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 or in lieu of Position c.4.b(1) and c.4.b(2), a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals (the provisions of SR 4.0.2 are not applicable); and A02

~~d.~~ ~~Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirement of any TS.~~ LA01

5.5.6 Inservice Testing Program

5.5.6 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components including applicable supports~~ pumps and valves. The program shall include the following: LA01

5.5.6.a a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.6 4.0.5 (Continued)

5.5.6.a b. Testing Frequencies applicable to the ASME OM Code and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5.6.b c. The provisions of SR 4.0.2 are applicable to the above required Frequencies and other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

5.5.6.c d. The provisions of SR 4.0.3 are applicable to inservice testing and activities; and

5.5.6.~~d~~ e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

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ITS A01 ITS 5.5

CONTAINMENT SYSTEMS

EMERGENCY GAS TREATMENT SYSTEM - EGTS - CLEANUP SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 Two independent emergency gas treatment system cleanup subsystems (EGTS) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. See ITS 3.6.10

ACTION:

With one EGTS cleanup subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8 Each EGTS cleanup subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on. See ITS 3.6.10

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Position C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 4000 cfm $\pm$ 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.c 3. Verifying a system flow rate of 4000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

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CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

LA02

5.5.9 5.5.9.c c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.9.d d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%.

2. Verifying that the filter train starts on a Phase A containment isolation Test Signal.

3. Verify the operation of the filter cooling bypass valves.

See ITS 3.6.10

4. Verifying that each system produces a negative pressure of greater than or equal to 0.5 inches W. G. in the annulus within 1 minute after a start signal.

See ITS 3.6.7

5.5.9 5.5.9.a e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

5.5.9 5.5.9.b f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 4000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.d 3. Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

LA02

5.5.9 d. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of representative carbon sample obtained in accordance
5.5.9.c with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9 e. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%.

2. Verifying that on a safety injection signal or a high radiation signal from the air intake stream, the system automatically diverts its inlet flow through the HEPA filters and charcoal adsorber banks. See ITS 3.7.10

5.5.9 f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-
5.5.9.a place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

5.5.9 g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon
5.5.9.b refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

h. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program. See ITS 3.7.10

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

PLANT SYSTEMS

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent auxiliary building gas treatment filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one auxiliary building gas treatment filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

4.7.8 Each auxiliary building gas treatment filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 10 hours with the heaters on.

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 9000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.d 3. Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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ITS A01 ITS 5.5

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of representative carbon sample obtained in accordance LA02

5.5.9.c with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9 d. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the filter train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the filter trains start on a Containment Phase A Isolation test signal.

3. Verifying that the system maintains the spent fuel storage area and the ESF pump rooms at a pressure equal to or more negative than minus 1/4 inch water gage relative the outside atmosphere while maintaining a total system flow of 9000 cfm ± 10%. See ITS 3.7.12

5.5.9.e 4. Verifying that the heaters dissipate 32 ± 3.2 kw when tested in accordance with ANSI N510-1975.

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-

5.5.9.a place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

5.5.9 f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon

5.5.9.b refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

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TABLE 4.8.1a

DIESEL GENERATOR BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A(1)	CATEGORY B(2)	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
5.5.15.b.2 Float Voltage	≥ 2.13 volts	≥ 2.13 volts(C)	> 2.07 volts
Specific Gravity(a)	≥ 1.195(b)	≥ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells > 1.190(b)

See ITS 3.8.6

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps.
~~(c) Corrected for average electrolyte temperature.~~ LA05
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

See ITS 3.8.6

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A01

ITS 5.5

TABLE 4.8.2

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A(1)	CATEGORY B(2)	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts (c)	> 2.07 volts
Specific Gravity(a)	≥ 1.200(b)	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells > 1.195(b)

See ITS 3.8.6

5.5.15.b.2

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps.
~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

See ITS 3.8.6

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REFUELING OPERATIONS

3/4.9.12 AUXILIARY BUILDING GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 One auxiliary building gas treatment filter train shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no auxiliary building gas treatment filter train OPERABLE, suspend all operations involving movement of fuel within the spent fuel pit or crane operation with loads over the spent fuel pit until at least one auxiliary building gas treatment filter train is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable. See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

4.9.12 The above required auxiliary buildings gas treatment filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 9000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.d 3. Verifying a system flow rate of 9000 cfm ± 10% during system operations when tested in accordance with ANSI N510-1975.

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ITS A01 ITS 5.5

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ LA02 that a laboratory analysis of representative carbon sample obtained in accordance with
5.5.9.c Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.9 d. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the filter train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the filter train starts on a high radiation signal from the fuel pool radiation monitoring system. See ITS 3.7.12

5.5.9.e 3. Verifying that the heaters dissipate 32 ± 3.2 kw when tested in accordance with ANSI N510-1975.

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter
5.5.9.a banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

5.5.9 f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the
5.5.9.b charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIMITING CONDITION FOR OPERATION

~~3.11.1.1 This specification is deleted.~~

~~3.11.1.2 This specification is deleted.~~

~~3.11.1.3 This specification is deleted.~~

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10
5.5.10.c 3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited ~~by the following expression:~~ LA03

$$\sum_i \frac{\text{concentration of isotope } i}{(\text{effluent concentration limit of isotope } i)} \le 6{,}700$$

~~excluding tritium and dissolved or entrained noble gases.~~

less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. A05

a. Condensate Storage Tank
b. Steam Generator Layup Tank
c. Outside temporary tanks for radioactive liquid

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA03

SURVEILLANCE REQUIREMENTS

5.5.10.c 4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days when radioactive materials are being added to the tank~~. LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. A04

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

LIMITING CONDITION FOR OPERATION

~~3.11.2.1 This specification is deleted.~~

~~3.11.2.2 This specification is deleted.~~

~~3.11.2.3 This specification is deleted.~~

~~3.11.2.4 This specification is deleted.~~

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10 5.5.10.a 3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the concentration of oxygen in a waste gas holdup tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in a waste gas holdup tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, without delay suspend all additions of waste gases to the affected waste gas holdup tank and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

LA03

SURVEILLANCE REQUIREMENTS

5.5.10.a 4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits ~~by monitoring the waste gas additions to the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.~~ LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Tank Monitoring Program Surveillance Frequencies. A04

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10
5.5.10.b 3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited ~~to less than or equal to 50,000 curies of noble gases (considered as Xe-133)~~.

to less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA03

SURVEILLANCE REQUIREMENTS

5.5.10.b 4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit ~~at least once per 24 hours when radioactive materials are being added to the tank.~~ LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Tank Monitoring Program Surveillance Frequencies. A04

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RADIOACTIVE EFFLUENTS

3/4.11.3 DELETED

LIMITING CONDITION FOR OPERATION

~~3.11.3 This specification is deleted.~~

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RADIOACTIVE EFFLUENTS

3/4.11.4 DELETED

LIMITING CONDITION FOR OPERATION

~~3.11.4 This specification is deleted.~~

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

~~3.12.1 This Specification is deleted.~~

~~3.12.2 This Specification is deleted.~~

~~3.12.3 This Specification is deleted.~~

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b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. DELETED

e. DELETED

f. Fire Protection Program implementation.

g. DELETED

See ITS 5.4

h. Quality Assurance Program for effluent and environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977, or Regulatory Guide 1.21, Rev. 1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 DELETED

6.8.3 DELETED

5.5 6.8.4 The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

5.5.2 A program to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The

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5.5.2 systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system at lease once per 18 months.

5.5.2 The provisions of SR 4.0.2 are applicable.

~~b. In-Plant Radiation Monitoring (DELETED)~~

5.5.8 c. Secondary Water Chemistry

5.5.8 A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.8.a (i) Identification of a sampling schedule for the critical variables and control points for these variables,

5.5.8.b (ii) Identification of the procedures used to measure the values of the critical variables,

5.5.8.c (iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

5.5.8.d (iv) Procedures for the recording and management of data,

5.5.8.e (v) Procedures defining corrective actions for off-control point chemistry conditions,

5.5.8.f (vi) Procedures identifying (a) the authority responsible for the interpretation of the data; and (b) the sequence and timing of administrative events required to initiate corrective action.

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~~d.~~ ~~DELETED~~

~~e.~~ ~~DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

5.5.3 A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.3.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and set-point determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to, UNRESTRICTED AREAS conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

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5.5.3.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be in accordance with the following:

5.5.3.g.1 1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

5.5.3.g.2 2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

5.5.3.j 10) Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.5.3 The provisions of SR 4.0.2 and 4.0.3 are applicable to the radioactive effluent controls program surveillance frequency.

~~g. Radiological Environmental Monitoring Program (DELETED)~~

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A01

ITS ITS 5.5

5.5.14 h. Containment Leakage Rate Testing Program

5.5.14.a A program shall ~~be established to implement~~ establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. ~~Visual examination and testing, including test intervals and extensions,~~ This program shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions and the following:

5.5.14.a.1 BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (~~13.2~~ psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

SII STET

~~12.16~~

5.5.14.b The peak calculated containment internal pressure for the design basis loss of coolant accident~~, P_a, is 12.0 psig.~~ is less than the containment design pressure of 12 psig. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig.

~~11.33~~ A13 ~~L03~~

5.5.14.c The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

5.5.14.d Leakage rate acceptance criteria are:

5.5.14.d.1 a. Containment overall leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests;

5.5.14.d.2 b. Air lock testing acceptance criteria are:

5.5.14.d.2 1) 1. Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.

5.5.14.d.2 2) 2. For each door, leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 6 psig for at least two minutes.

5.5.14.d.3 c. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate less than or equal to 0.05 L_a.

5.5.14.d.4 d. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING acceptance criteria are:

5.5.14.d.4 1) 1. The combined bypass leakage rate to the auxiliary building shall be less than or equal to 0.25 L_a by applicable Type B and C tests.

5.5.14.d.4 2) 2. Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (~~12~~ psig) during each Type A test.

~~11.33~~

5.5.14.f The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

5.5.14.e The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

~~i. Configuration Risk Management Program (DELETED)~~

5.5.12 j. Technical Specification (TS) Bases Control Program

5.5.12 This program provides a means for processing changes to the Bases of TSs.

5.5.12.a a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.12.b b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.12.b.1 1. A change in the TS incorporated in the license or

5.5.12.b.2 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.12.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.5.12.d d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.7 k. Steam Generator (SG) Program

5.5.7 A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

5.5.7.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected and/or plugged, to confirm that the performance criteria are being met.

5.5.7.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

5.5.7.b.1 1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full

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5.5.7.b.1 power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5.7.b.2 2. Accident induced leakage performance criterion: The accident-induced leakage is not to exceed 1.0 gpm for the faulted SG. The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the maximum leakage rate established in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG.

5.5.7.b.3 3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System, Operational Leakage."

A06

5.5.7.c c. Provisions for SG tube ~~repair~~ plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

5.5.7.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube ~~repair~~ plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5.7.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG ~~replacement.~~ installation A06

5.5.7.d.2 2. INSERT 1 L01 ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period.~~ No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

L01

INSERT 1

After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

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affected and potentially affected A07

5.5.7.d.3 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever ~~is less~~). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

results in more frequent inspections A07

5.5.7.e e. Provisions for monitoring operational primary-to-secondary leakage.

5.5.4 l. Component Cyclic and Transient Limit

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 DELETED

6.9.1.2 DELETED

6.9.1.3 DELETED

ANNUAL REPORTS [1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality. See ITS 5.6

6.9.1.5 DELETED

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

~~6.13 PROCESS CONTROL PROGRAM (PCP) (DELETED)~~

5.5.1 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1 6.14.1 Changes to the ODCM:

5.5.1.a 1. Shall be documented and records of reviews performed shall be retained in a manner convenient for review. This documentation shall contain:

5.5.1.a.1 a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b. A determination that the change will maintain the level of radioactive effluent control pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

plant manager M01

5.5.1.b 2. Shall become effective after review and acceptance by the ~~process described in TVA-NQA-PLN89-A~~. LA04

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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~~6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)** (DELETED)~~

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5.5.11 ## 6.16 DIESEL FUEL OIL TESTING PROGRAM

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

5.5.11.a a. Acceptability of new fuel oil prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color; or a water and sediment content within limits L02

5.5.11.b b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

5.5.11.c c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days ~~in accordance with ASTM D-2276, Method A~~ ~~ASTM D6217-11~~. LA06 A12

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing A08

5.5.16 ## 6.17 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

5.5.16.a a. The definition of the CRE and the CRE boundary.

5.5.16.b b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

5.5.16.c c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

5.5.16.d d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP ~~of 4000 cubic feet per minute plus or minus 10 percent~~, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary. A09 A10

5.5.16 6.17 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (continued)

5.5.16.e e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

5.5.16.f f. The provisions of SR 4.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.13 Add proposed program 5.5.13 M02

5.5.15 Add proposed program 5.5.15 M03

5.5.17 Add proposed program 5.5.17 M04

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IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or See ITS 1.0

c. Reactor coolant system leakage through a steam generator to the secondary system (primary to secondary leakage).

MEMBER(S) OF THE PUBLIC

1.17 DELETED

OFFSITE DOSE CALCULATION MANUAL

5.5.1.a 1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program.

5.5.1.b The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, or component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). See ITS 1.0

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.3 (Continued)

up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be entered. See ITS 3.0

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

5.5.6 4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1, 2 and 3 ~~components~~ pumps and valves shall be as follows: LA01

~~Inservice Inspection Program~~

~~This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:~~

~~a. Provisions that inservice testing of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;~~ LA01

~~b. The provisions of SR 4.0.2 are applicable to the frequencies for performing inservice inspection activities;~~

5.5.5 c. Inspection of each reactor coolant pump flywheel per the recommendation of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 or in lieu of Position c.4.b(1) and c.4.b(2), a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals (the provisions of SR 4.0.2 are not applicable); and A02

~~d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirement of any TS.~~ LA01

5.5.6 Inservice Testing Program

5.5.6 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves ~~components including applicable supports~~. The program shall include the following: LA01

5.5.6.a

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.6 4.0.5 (Continued)

5.5.6.a a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies applicable to the ASME OM Code and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5.6.b c. The provisions of SR 4.0.2 are applicable to the above required Frequencies and other normal and accelerated frequencies specified as 2 years or less in the Inservice Test Program for performing inservice testing activities;

5.5.6.c d. The provisions of SR 4.0.3 are applicable to inservice testing and activities; and

5.5.6.d e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

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CONTAINMENT SYSTEMS

EMERGENCY GAS TREATMENT SYSTEM - EGTS - CLEANUP SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 Two independent emergency gas treatment system cleanup subsystems (EGTS) shall be OPERABLE.

See ITS 3.6.10

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one EGTS cleanup subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8 Each EGTS cleanup subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on. See ITS 3.6.10

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Position C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 4000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.c 3. Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

LA02

5.5.9 5.5.9.c c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.d d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm + 10%.

2. Verifying that the filter train starts on a Phase A containment isolation Test Signal.

3. Verify the operation of the filter cooling bypass valves.

See ITS 3.6.10

4. Verifying that each system produces a negative pressure of greater than or equal to 0.5 inches W.G. in the annulus within 1 minute after a start signal.

See ITS 3.6.7

5.5.9 5.5.9.a e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm + 10%.

5.5.9 5.5.9.b f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm + 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 4000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying~~, within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.d 3. Verifying a system flow rate of 4000 cfm + 10% during system operation when tested in accordance with ANSI N510-1975.

LA02

5.5.9 d. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with
5.5.9.c Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ATSM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.9 e. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%.

2. Verifying that on a safety injection signal or high radiation signal from the air intake stream, the system automatically diverts its inlet flow through the HEPA filters and charcoal adsorber banks. See ITS 3.7.10

5.5.9 f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter
5.5.9.a banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

5.5.9 g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon
5.5.9.b refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

h. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program. See ITS 3.7.10

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

PLANT SYSTEMS

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent auxiliary building gas treatment filter trains shall be OPERABLE.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

With one auxiliary building gas treatment filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

4.7.8 Each auxiliary building gas treatment filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 10 hours with the heaters on.

Add proposed ITS 5.5.9 generic program statement A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

5.5.9.a 5.5.9.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 9000 cfm ± 10%.

LA02

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.a 5.5.9.b 5.5.9.d 3. Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ LA02
5.5.9.c that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9 d. At least once per 18 months by:

5.5.9.d
1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the filter train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the filter trains start on a Containment Phase A Isolation test signal.

3. Verifying that the system maintains the spent fuel storage area and the ESF pump rooms at a pressure equal to or more negative than minus 1/4 inch water gauge relative the outside atmosphere while maintaining a total system flow of 9000 cfm ± 10%. See ITS 3.7.12

5.5.9.e
4. Verifying that the heaters dissipate 32 ± 3.2 kw when tested in accordance with ANSI N510-1975.

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place
5.5.9.a in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

5.5.9 f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon
5.5.9.b refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A03

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TABLE 4.8-1a

DIESEL GENERATOR BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEROGY A[(1)]	CATEGORY B[(2)]	
	Limit for each designated pilot cell	Limits for each connected cell	Allowable[(3)] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[(c)]	> 2.07 volts
Specific Gravity[(a)]	≥ 1.195[(b)]	≥ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190[(b)]

See ITS 3.8.6

5.5.15.b.2

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps.

~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

See ITS 3.8.6

ITS

A01

ITS 5.5

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEROGY A[1]	CATEGORY B[2]	
	Limit for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[b]

See ITS 3.8.6

5.5.15.b.2

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps.
~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

See ITS 3.8.6

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ITS 5.5

REFUELING OPERATIONS

3/4.9.12 AUXILIARY BUILDING GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 One auxiliary building gas treatment filter train shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no auxiliary building gas treatment filter train OPERABLE, suspend all operations involving movement of fuel within the spent fuel pit or crane operation with loads over the spent fuel pit until at least one auxiliary building gas treatment filter train is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

4.9.12 The above required auxiliary building gas treatment filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

Add proposed ITS 5.5.9 generic program statement

A03

5.5.9 b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory
5.5.9.a
5.5.9.b
Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 9000 cfm ± 10%.

LA02

2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory
5.5.9.c
Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

3. Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

5.5.9.a
5.5.9.b
5.5.9.d

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A01

ITS 5.5

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

LA02

5.5.9 c. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86° F) and a relative humidity of 70%.

5.5.9.c

5.5.9 d. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3 inches Water Gauge while operating the filter train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the filter train starts on a high radiation signal from the fuel pool radiation monitoring system. See ITS 3.7.12

5.5.9.e 3. Verifying that the heaters dissipate 32 ± 3.2 kw when tested in accordance with ANSI N510-1975.

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

5.5.9.a

5.5.9 f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

5.5.9.b

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

A03

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIMITING CONDITION FOR OPERATION

~~3.11.1.1 This Specification is deleted.~~

~~3.11.1.2 This Specification is deleted.~~

~~3.11.1.3 This Specification is deleted.~~

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10
5.5.10.c 3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited ~~by the following expression:~~ LA03

$$\sum_i \frac{\text{concentration of isotope } i}{\text{(effluent concentration limit of isotope } i)} \leq 6{,}700$$

~~excluding tritium and dissolved or entrained noble gases.~~

less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. A05

a. Condensate Storage Tank
b. Steam Generator Layup Tank
c. Outside temporary tanks for radioactive liquid

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA03

SURVEILLANCE REQUIREMENTS

5.5.10.c 4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days when radioactive materials are being added to the tank~~. LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. A04

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

LIMITING CONDITION FOR OPERATION

~~3.11.2.1 This Specification is deleted.~~

~~3.11.2.2 This Specification is deleted.~~

~~3.11.2.3 This Specification is deleted.~~

~~3.11.2.4 This Specification is deleted.~~

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10 5.5.10.a 3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume~~.

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the concentration of oxygen in a waste gas holdup tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in a waste gas holdup tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, without delay suspend all additions of waste gases to the affected waste gas holdup tank and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~ LA03

SURVEILLANCE REQUIREMENTS

5.5.10.a 4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits ~~by monitoring the waste gas additions to the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10~~. LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Tank Monitoring Program Surveillance Frequencies. A04

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ITS A01 ITS 5.5

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement A04

5.5.10
5.5.10.b 3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited ~~to less than or equal to 50,000 curies of noble gases (considered as Xe-133)~~.

to less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

~~APPLICABILITY: At all times.~~

~~ACTION~~:

~~a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA03

SURVEILLANCE REQUIREMENTS

5.5.10.b 4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit ~~at least once per 24 hours when radioactive materials are being added to the tank~~. LA03

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Tank Monitoring Program Surveillance Frequencies. A04

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RADIOACTIVE EFFLUENTS

3/4.11.3 DELETED

LIMITING CONDITION FOR OPERATION

~~3.11.3 This Specification is deleted.~~

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RADIOACTIVE EFFLUENTS

3/4.11.4 DELETED

LIMITING CONDITION FOR OPERATION

~~3.11.4 This Specification is deleted.~~

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

~~3.12.1 This Specification is deleted.~~

~~3.12.2 This Specification is deleted.~~

~~3.12.3 This Specification is deleted.~~

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h. Quality Assurance Program for effluent and environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21, Rev. 1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 DELETED

6.8.3 DELETED

See ITS 5.4

5.5 6.8.4 The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

5.5.2 A program to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system at lease once per 18 months.

5.5.2 The provisions of SR 4.0.2 are applicable

~~b. In-Plant Radiation Monitoring (DELETED)~~

5.5.8 c. Secondary Water Chemistry

5.5.8 A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.8.a (i) Identification of a sampling schedule for the critical variables and control points for these variables,

5.5.8.b (ii) Identification of the procedures used to measure the values of the critical variables,

5.5.8.c (iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage

5.5.8.d (iv) Procedures for the recording and management of data,

5.5.8.e (v) Procedures defining corrective actions for off-control point chemistry conditions,

5.5.8.f (vi) Procedures identifying (a) the authority responsible for the interpretation of the data; and (b) the sequence and timing of administrative events required to initiate corrective action.

~~d. Deleted~~

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~~e. DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

5.5.3 A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.3.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and set-point determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to, UNRESTRICTED AREAS conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

5.5.3.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases

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6.8.4 f. Radioactive Effluent Controls Program (Cont.)

5.5.3.f of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be in accordance with the following:

5.5.3.g.1 1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

5.5.3.g.2 2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

5.5.3.j 10) Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.5.3 The provisions of SR 4.0.2 and 4.0.3 are applicable to the radioactive effluent controls program surveillance frequency.

~~g. Radiological Environmental Monitoring Program (DELETED)~~

5.5.14 h. Containment Leakage Rate Testing Program

establish

5.5.14.a A program shall ~~be established to implement~~ the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. ~~Visual examination and testing, including test intervals and extensions,~~ This program shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions and the following:

5.5.14.a.1 BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (~~13.2~~ psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

STET

~~12.46~~

SII

5.5.14.b The peak calculated containment internal pressure for the design basis loss of coolant accident, ~~P_a, is 12.0 psig.~~ is less than the containment design pressure of 12 psig. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig.

~~11.33~~

A13 ~~L03~~

5.5.14.c The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

5.5.14.d Leakage rate acceptance criteria are:

5.5.14.d.1 a. Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

5.5.14.d.2 b. Air lock testing acceptance criteria are:

5.5.14.d.2 1) 1)Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

5.5.14.d.2 2) 2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 6 psig for at least two minutes.

5.5.14.d.3 c. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate less than or equal to $0.05\ L_a$.

5.5.14.d.4 d. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING acceptance criteria are:

5.5.14.d.4 1) 1. The combined bypass leakage rate to the auxiliary building shall be less than or equal to $0.25\ L_a$ by applicable Type B and C tests.

5.5.14.d.4 2) SII 2. Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (~~12 psig~~) during each Type A test. STET ~~11-33~~ ~~L03~~

5.5.14.f The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

5.5.14.e The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

~~i. Configuration Risk Management Program (DELETED)~~

5.5.12 j. Technical Specification (TS) Bases Control Program

5.5.12 This program provides a means for processing changes to the Bases of these TSs.

5.5.12.a a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.12.b b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.12.b.1 1. A change in the TS incorporated in the license or

5.5.12.b.2 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.12.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.5.12.d d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.7 k. Steam Generator (SG) Program

5.5.7 A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

5.5.7.a a. Provisions for Condition Monitoring Assessments.

Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

5.5.7.b b. Provisions for Performance Criteria for SG Tube Integrity.

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

5.5.7.b.1 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents (DBAs). This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the DBA primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the DBAs, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5.7.b.2 2. Accident induced leakage performance criterion: The accident-induced leakage is not to exceed 1.0 gpm for the faulted SG ~~and 0.1 gpm for each of the non-faulted SGs~~. The primary-to-secondary accident induced leakage rate for any DBA, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. A11

5.5.7.b.3 3. The operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, “Reactor Coolant System, Operational Leakage.”

5.5.7.c c. Provisions for SG Tube ~~Repair~~ plugging Criteria. A06

Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

5.5.7.d d. Provisions for SG Tube Inspections.

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube ~~repair~~ plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5.7.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG ~~replacement.~~ installation A06

5.5.7.d.2 2. ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72 and~~ INSERT 1 ~~thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period.~~ No SGs shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected. L01

5.5.7.d.3 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever ~~is less~~ results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. A07 A07

5.5.7.e e. Provisions for Monitoring Operational Primary-to-Secondary Leakage.

5.5.4 l. Component Cyclic and Transient Limit

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

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L01

INSERT 1

After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

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ADMINISTRATIVE CONTROLS

~~6.13 PROCESS CONTROL PROGRAM (PCP) (DELETED)~~

5.5.1 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1 6.14.1 Changes to the ODCM:

5.5.1.a 1. Shall be documented and records of reviews performed shall be retained in a manner convenient for review. This documentation shall contain:

5.5.1.a.1 a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b 2. Shall become effective after review and acceptance by the plant manager ~~process described in TVA-NQA-PLN89-A~~. M01 LA04

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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~~6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)** (DELETED)~~

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5.5.11 6.16 DIESEL FUEL OIL TESTING PROGRAM

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

5.5.11.a a. Acceptability of new fuel oil prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color; or a water and sediment content within limits L02

5.5.11.b b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

5.5.11.c c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days ~~in accordance with ASTM D-2276, Method A~~. ~~ASTM D6217-11~~ LA06 A12

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing A08

5.5.16 6.17 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

5.5.16.a a. The definition of the CRE and the CRE boundary.

5.5.16.b b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

5.5.16.c c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

5.5.16.d d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP ~~of 4000 cubic feet per minute plus or minus 10 percent~~, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary. A09 A10

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5.5.16 6.17 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (continued)

5.5.16.e e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

5.5.16.f f. The provisions of SR 4.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.13 Add proposed program 5.5.13 M02

5.5.15 Add proposed program 5.5.15 M03

5.5.17 Add proposed program 5.5.17 M04

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DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 4.0.5.c states in part, that the provisions of CTS SR 4.0.2 are not applicable for the 20 year interval reactor coolant pump flywheel inspection. ITS 5.5.5 requires a program to provide for the inspection of each reactor coolant pump flywheel. This changes the CTS by not stating that the allowance of ITS SR 3.0.2 is not applicable.

This change is acceptable because no changes have been made to the existing requirements. The CTS and proposed ITS 5.5.5 continue to require the same reactor coolant pump flywheel inspections to be performed. A statement that ITS SR 3.0.2 is not applicable is not needed, as the provisions of SR 3.0.2 do not apply to the programs in ITS Section 5.5, unless specified. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 The Surveillances associated with the ventilation filter testing for the Control Room Ventilation System (CREVS), the Emergency Gas Treatment System (EGTS), and the Auxiliary Building Gas Treatment System (ABGTS) have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.9). As such, a general program statement has been added as ITS 5.5.9. Also, a statement of the applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply (as allowed in the CTS). This changes the CTS by moving the ventilation filter testing Surveillances associated with the CREVS, EGTS, and ABGTS to a program in ITS 5.5 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Surveillances. The addition of the ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the CTS, therefore, it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 The liquid holdup tank requirements in CTS 3.11.1.4, the explosive gas mixture requirements of CTS 3.11.2.5, and the gas decay tanks requirements in CTS 3.11.2.6 have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.10). As such, a general program statement has been added. Also, a statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify the allowances for Surveillance Frequency extensions do apply. This changes the CTS by moving the liquid holdup tank, the explosive gas mixture, and the gas decay tanks requirements to a program in ITS 5.5.10 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

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The addition of the program statement is acceptable because it is describing the intent of the CTS Specification. The addition of ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 3.11.1.4 requires that the quantity of radioactive material contained in the condensate storage tank, steam generator layup tank and outside temporary tanks for radioactive liquid shall be less than or equal to 6700 effluent concentration limit (ECL). CTS 4.11.1.4 requires a determination that the radioactive material contained in each of the tanks listed in CTS 3.11.1.4 is within limits on a prescribed frequency. ITS 5.5.10.c requires a surveillance program to ensure that the quantity of radioactive material contained in all outdoor temporary liquid radwaste storage tanks, Condensate Storage tank, and Steam Generator Layup tank is less than the amount that would result in concentrations exceeding the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. This changes the CTS by specifically stating that the program shall meet the 10 CFR 20 requirements (See DOC LA03 for discussion of the removal of the effluent concentration limit).

The addition of the 10 CFR 20 limitations is acceptable because 10 CFR 20.1002 states that this part applies to persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility under parts 30 through 36, 39, 40, 50, 52, 60, 61, 63, 70, or 72 of this chapter. SQN Units 1 and 2 are licensed by the Nuclear Regulatory Commission, in part, under 10 CFR Parts 30, 40, 50, and 70. 10 CFR 20.1302 requires, in part, that the annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area to not exceed the values specified in table 2 of appendix B to part 20. 10 CFR 20, Appendix B, Table 2 refers to effluent concentrations and Column 2 of this table lists limitations associated with water (liquid). Therefore, SQN Units 1 and 2 are currently required to limit effluent releases to within these concentrations. Additionally, restricting the quantity to less than or equal to 6700 ECL (See DOC LA03) provides assurance that the resulting concentrations would be less than the limits of 10 CFR 20. This change is designated as administrative because it does not result in a technical change to the CTS.

A06 CTS 6.8.4.k states the requirements of the Steam Generator (SG) program. ITS 5.5.7 specifies the requirements of the Steam Generator (SG) program based on the latest revision of TSTF-510. This changes CTS 6.8.4.k.c and CTS 6.8.4.k.d by replacing the word "repair" with "plugging" and replacing the word "replacement" with "installation." CTS 6.8.4.k.d.2 has been revised to reflect TSTF-510-A.

This change is acceptable because no changes have been made to the existing requirements. ITS 5.5.7 continues to require the same steam generator inspections to be performed in accordance with approved TSTF-510-A. This change is designated as administrative because it does not result in technical changes to the CTS.

A07 The first sentence of CTS 6.8.4.k.d.3 states, "If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less)." The first sentence of ITS 5.5.7.d.3 states, "If crack

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indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)." The proposed change is replacing the words "for each SG" with the words "for each affected and potentially affected SG," and is replacing the parenthetical statement "(whichever is less)" with "(whichever results in more frequent inspections)".

The purpose of CTS 6.8.4.k.d.3 is to restrict the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less) once cracks have been found in any SG tube. The intent of this requirement is that it applies to the affected SG and to any other SG which may be and affected by the degradation mechanism that caused the known crack(s). This change was made to reflect changes made under TSTF-510 and is acceptable because it clarifies the intent of the paragraph. This change is designated as administrative because it does not result in a technical change to the CTS.

A08 The Diesel fuel oil testing program (CTS 6.16) has been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.11). As such, a statement of the applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extension do apply. This changes the CTS by specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A09 CTS 6.17.d requires, in part, that one train of the Control Room Emergency Ventilation System (CREVS) operates at a flow rate of 4000 cubic feet per minute plus or minus 10 percent. ITS 5.5.16.d requires, in part that one train of the CREVS operates at the flow rate required by the Ventilation Filter Testing Program (VFTP). This changes the CTS by requiring the CREVS to operate at the flow rate required by the VFTP.

The change is acceptable because no change to the existing requirements have been made. ITS 5.5.9 contains the flow requirements for a OPERABLE CREVS train. This change is designated as administrative because it does not result in a technical change to CTS.

A10 CTS 6.17.d requires, in part, measurement of the Control Room Envelope (CRE) boundary be tested using one train of the Control Room Emergency Ventilation System (CREVS) every 36 months on a STAGGERED TEST BASIS. CTS 1.35 defines a STAGGERED TEST BASIS as, "a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval." ITS 5.5.16.d requires a similar test of the CRE boundary with use of one CREVS train every 18 months "on a STAGGERED TEST BASIS." In ITS, a STAGGERED TEST BASIS consists of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals,

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where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. This changes the CTS by utilizing the ITS definition of a STAGGERED TEST BASIS.

This change is acceptable because the requirements for CRE boundary testing remain unchanged. The ITS definition of STAGGERED TEST BASIS and its application in this requirement do not change the CTS 6.17.d testing Frequency requirements. CTS 6.17.d requires each train of CREVS to be tested at least once per 36 months (one train each 18 months). ITS 5.5.16.d requires a train of CREVS to be tested each 18 months, alternating between the trains each interval. Therefore, the CTS and ITS testing Frequencies are the same. This change is designated as administrative because it does not result in technical changes to the CTS.

A11 SQN Unit 2 CTS 6.8.4.k.b.2, Steam Generator (SG) Program – Accident Induced Leakage Performance Criterion, states, in part, that the accident-induced leakage is not to exceed 1.0 gpm for the faulted SG and 0.1 gpm for each of the non-faulted SGs. Both Unit 1 and Unit 2 CTS 6.8.4.k.b.3 contain criterion for operational leakage referencing the CTS 3.4.6.2 criterion of a maximum primary to secondary leakage of 150 gallons per day (gpd) through any one steam generator. ITS 5.5.7.b.2, Steam Generator (SG) Program – Accident Induced Leakage Performance Criterion, states, in part, that Leakage is not to exceed 1 gpm per SG while ITS 5.5.7.b.3 states that the operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE," 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG). This changes the CTS by removing the duplicative non-faulted SG leakage criterion.

The purpose of CTS 6.8.4.k.b is to provide provisions for performance criteria for SG tube integrity. The provisions are provided such that SG tube integrity is maintained by meeting performance criteria for tube structural integrity, accident induced leakage, and operational leakage. CTS 6.8.4.k.b.2 for Unit 1 provides one criterion for accident induced leakage, 1 gpm for the faulted SG; whereas CTS 6.8.4.k.b.2 for Unit 2 provides two criterion for accident induced leakage, 1.0 gpm for the faulted SG and 0.1 gpm for each of the non-faulted SGs. Both Unit 1 and Unit 2 CTS 6.8.4.k.b.3 provide an operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage." The CTS 3.4.6.2 requirement for the maximum primary to secondary leakage is 150 gallons per day (gpd) (0.1 gpm) through any one steam generator. The Unit 2 CTS 6.8.4.k.b.2 criterion of 0.1 gpm for each of the non-faulted SGs is duplicated in the Unit 2 CTS 6.8.4.k.b.3 criterion of the operational leakage performance criterion as reference to Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage," of 150 gpd through any one SG. This change is acceptable because duplicative leakage criterion from the Unit 2 CTS is being removed while the leakage criterion is being maintained. This change is designated as administrative because it does not result in a technical change to the CTS.

A12 CTS 6.16 c requires the total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A. ITS 5.5.11 c requires that the total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days but does not include a specific test method. TVA is proposing to change the test method for determining total particulate concentration for SQN to ASTM D6217-

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11. This changes the CTS by requiring the testing of fuel oil total particulate concentration to be in accordance with ASTM D6217-11.

The purpose of CTS 6.16 c is to provide the requirements for testing of total particulate concentration of the fuel oil. Regulatory Guide 1.137, Revision 2, "Fuel Oil Systems for Emergency Power Supplies," describes methods that the NRC considers acceptable for use in complying with the NRC requirements regarding fuel oil systems. Based on the guidance of Regulatory Guide 1.137 ANSI/ANS-59.51 to ANSI/ASTM D2276-94 for manual sampling of the stored fuel should be changed to ASTM D6217-11. This change is acceptable because testing of total particulate concentration of the fuel oil will be done in accordance with the approved NRC method of ASTM D6217-11. This change is designated as administrative because it does not result in a technical change to the CTS.

SII

INSERT 1

MORE RESTRICTIVE CHANGES

M01 CTS 6.14.1.2 states, in part, that the ODCM becomes effective after review and acceptance by the process described in TVA-NQA-PLN89-A. ITS 5.5.1.c.2 states, in part, that the ODCM becomes effective after review and acceptance by the plant manager. This changes the CTS by requiring the plant manager approval for the ODCM.

The purpose of CTS 6.14.1.2 is to ensure that the ODCM has been properly reviewed by the process described in TVA-NQA-PLN89-A. ITS 5.5.1 still requires that the review process described in TVA-NQA-PLN89-A is performed (see DOC LA04 for the exclusion of the process described in TVA-NQA-PLN89-A from the ITS 5.5.1), but also includes an additional acceptance that the plant manager must review and approve the ODCM. This change is designated as more restrictive since a higher level of approval is required in the ITS than was required in the CTS.

M02 The CTS does not include program requirements for the Safety Function Determination Program. The ITS includes a program for the Safety Function Determination Program. This change the CTS by adding the Safety Function Determination Program (SFDP).

The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. The specific wording associated with this program is found in ITS 5.5.13. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M03 The CTS does not include a requirement for the Battery Monitoring and Maintenance Program. The ITS includes a requirement for this program. This changes the CTS by adding the ITS 5.5.15, "Battery Monitoring and Maintenance Program."

The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance. The specific wording associated with this program may be found in ITS 5.5.15. This change is acceptable because it supports implementation of the requirements of the ITS. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

INSERT 1

SII

A13 CTS 6.8.4.h states, in part, "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig." ITS 5.5.14.b states, "The calculated peak containment internal pressure for the design basis loss of coolant accident is less than the containment design pressure of 12 psig. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig." This changes the CTS by stating the peak calculated containment internal pressure for the design basis loss of coolant accident is less than the design pressure of 12 psig, and specifying that for the Containment Leakage Rate Testing Program P_a is defined as 12.0 psig.

This change is acceptable because no changes have been made to the existing requirements. ITS 5.5.14 continues to use the value of 12.0 psig for the term P_a. The value for the calculated peak containment pressure for the deign basis loss of coolant accident is provided in the Applicable Safety Analyses Section of the Bases for ITS Specifications 3.6.4 and 3.6.6. This change is designated as administrative because it does not result in technical changes to the CTS.

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M04 The CTS does not have a Surveillance Frequency Control Program. ITS 5.5.17 requires a program to satisfy the relocation of the Surveillance Frequency from the individual specifications. This changes the CTS by incorporating the requirements of ITS 5.5.17.

The NRC has been reviewing and granting improvements to the Improved Standard Technical Specifications (ISTS) based, at least in part, on probabilistic risk analysis insights. Typically, the proposed improvements involved a relaxation of one or more Completion Times or Surveillance Frequencies in the TS. In August 1995, the NRC adopted a final policy statement on the use of probabilistic risk assessment (PRA) methods, which included the following regarding the expanded use of PRA.

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, licensee commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.
- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on need for proposing and backfitting new generic requirements on nuclear power plant licensees.

In its approval of the policy statement, the Commission articulated its expectation that implementation of the policy statement will improve the regulatory process in three areas: foremost, through safety decision-making enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees. This change is consistent with TSTF-425-A. TSTF- 425-A required that licensees who adopted this TSTF confirm that the plant PRA is consistent with Section 4.2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment results for Risk-Informed Activities." SQN has performed an assessment on the Sequoyah Units 1 and 2 PRA, and confirmed that it is consistent with the guidance in Section 4.2 of Regulatory Guide 1.200 (See Enclosure 10). Future model updates (internal model or external model) will be evaluated to determine any impact on the conclusions of the assessment that was performed in support of adopting this change. For each individual Surveillance Frequency relocation, see each of the associated Technical Specifications for the

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Discussion of Changes (DOC) justifying the individual relocations. This change is considered more restrictive since a new program is being added to the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 4.0.5 provides requirements for the Inservice Inspection Program. The ITS does not include Inservice Inspection Program requirements. In addition, since the Inservice Testing Program is the only requirement remaining, the reference to ASME Code Class 1, 2, and 3 "components" has been changed to "pumps and valves" for clarity. Pumps and valves are the only components related to the Inservice Testing Program (as described in CTS 4.0.5). This changes the CTS by moving these requirements from the Technical Specifications to the Inservice Inspection (ISI) Program.

The removal of these requirements is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications still retain requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled by the ISI, which is required by 10 CFR 50.55a. Compliance with 10 CFR 50.55a is required by the SQN Units 1 and 2 Operating Licenses. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS 4.6.1.8.b.2, CTS 4.6.1.8.c, CTS 4.7.7.c.2, CTS 4.7.7.d, CTS 4.7.8.b.2, CTS 4.7.8.c, CTS 4.9.12.b.2, CTS 4.9.12.c require that within 31 days after removal of a carbon sample the laboratory analysis results are shown to be within limit. ITS 5.5.9.c requires the same analysis to be performed however the detail of "within 31 days after removal of a carbon sample" is not included. This changes the CTS by moving these procedural details from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to perform the testing at the appropriate Frequencies. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

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LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.1.4 includes the details for implementing the requirements for the liquid holdup tank. CTS 3.11.2.5 includes the details for implementing the requirements for the explosive gas mixture. CTS 3.11.2.6 includes the details for implementing the requirements for the gas decay tanks. The details for implementing these requirements, including the specific limits, are not included in the ITS. CTS 3.11.2.6 Bases requires, in part in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion boundary will not exceed 0.5 rem. ITS 5.5.10.b requires the curie content in the gas decay tank to be less than the amount that would result in whole body exposure of greater than or equal to 0.5 rem at the exclusion boundary. This changes the CTS by moving the boundary exposure limit of 0.5 rem from the Bases to ITS 5.5.10 and moving those procedural details for implementing the requirements, including the specific limits, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for the specific limits, Applicability, Actions, and Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.10 still retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in condensate storage tank, steam generator layup tank and outdoor temporary liquid radwaste storage tanks. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, ISI Program)* CTS 6.14.1.2 requires changes to the ODCM to be effective after review and acceptance by the process described in TVA-NQA-PLN89-A. ITS 5.5.1.b requires changes to the ODCM to become effective after the approval of the plant manager. This changes the CTS by moving the process described in TVA-NQA-PLN89-A to the Nuclear Facility Quality Assurance Program Description (NFQAPD). DOC M01 describes the addition of the plant manager approval.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because these types of procedural details will be adequately controlled in the NFQAPD. Any changes to the NFQAPD are made under 10 CFR 50.54(a), which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specifications requirements are being removed from the Technical Specifications.

LA05 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS Table 4.8.1a and Table 4.8.2 Unit 1 footnote (c) and CTS Table 4.8-1a and Table 4.8-2 Unit 2 footnote (c) states, in part the float voltage of ≥ 2.13 volts

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is corrected for average electrolyte temperature. ITS 5.5.15 b.1 requires a program with actions to restore battery cells with float voltage < 2.13 V and ITS 5.5.15 b.2 requires a program with actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cells has been found to be < 2.13 V. This changes the CTS by by moving information from the specification to the Battery Monitoring and Maintenance Program implementing document.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.15 still retains the requirement for float voltage ≥ 2.13 V. Also, this change is acceptable because these types of procedural details will be adequately controlled by the requirements of a program required by ITS Chapter 5. ITS 5.5.15, Battery Monitoring and Maintenance program is controlled by Chapter 5 of the Technical Specifications. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA06 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 6.16 c requires total particulate of the fuel oil to be less than or equal to 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A. ITS 5.5.11.c requires the total particulate concentration of the fuel oil is less than or equal to 10 mg/l when tested every 31 days. This changes the CTS by moving the details of using of particulate testing standard ASTM D-2276, Method A from the CTS to the ITS SR 3.8.3.3 Bases.

The removal of these details related to testing standards from the Technical Specification is acceptable, because this type of information is not necessary to be included in the Technical Specification to provide adequate protection of the public health and safety. The ITS retains the requirement for fuel oil particulate testing every 31 days. Also, this change is acceptable because the removed details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as less restrictive removal of detail change, because information relating to testing standards is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 6.8.4.k.d.2 states, "Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." ITS 5.5.7.d.2 states, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of

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tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." ITS 5.5.7.d.2 goes on to describe the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d by stating, "a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period; b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods." This changes the CTS by relaxing the surveillance frequency for inspecting SG tubes.

The purpose of CTS 6.8.4.k is to ensure that SG tube integrity is maintained by providing provisions regarding the scope, frequency, and methods of SG tube inspections. These changes to when inspections are performed are considered marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections and reflect the improved resistance of alloy 690 TT SG tubes to stress corrosion cracking. This change is acceptable because TVA has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," (ADAMS Accession No. ML110610350) and the model safety evaluation dated October 19, 2011 (ADAMS Accession No. ML112101513) as part of the Federal Register Notice for Comment. As described in the subsequent paragraphs, TVA has concluded that the justifications presented in TSTF-510 and the model safety evaluation prepared by the NRC staff are applicable to SQN Unit 1 and Unit 2 and justify the incorporation of the changes to the SQN Unit 1 and SQN Unit 2 ITS. TVA is proposing the following variations from the TS changes described in the TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated October 19, 2011. SQN Unit 1 and Unit 2 ITS utilize different numbering than the Standard Technical Specifications (NUREG 1431, Revision 4.0, "Standard Technical Specifications Westinghouse Plants") on which TSTF-510 was based. The specific numbering differences are: 1) TSTF-510 Rev. 2, TS 3.4.20, "Steam Generator Tube Integrity," is ITS 3.4.17, "Steam Generator Tube Integrity"; 2) TSTF-510 Rev. 2 TS 5.5.9, "Steam Generator (SG) Program is ITS 5.5.7, "Steam Generator (SG) Program"; and 3) TSTF-510 Rev. 2 TS 5.6.7, "Steam Generator Tube Inspection Report," is ITS 5.6.6, "Steam Generator Tube Inspection Report." This change is designated as less restrictive because the SG tube inspections will be performed less frequently in ITS than they were in CTS.

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L02 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 6.16.a.3 requires performance of the "clear and bright" test, used to establish the acceptability of new fuel oil for use prior to addition to storage tanks. ITS 5.5.11.a.3 requires a determination that the fuel oil has a clear and bright appearance with proper color or that water and sediment content is within limits. This changes the CTS by allowing a "water and sediment content" test to be performed to establish the acceptability of new fuel oil instead of only allowing a "clear and bright" test.

CTS 6.16.a.3 requires performance of the "clear and bright" test, to establish the acceptability of new fuel oil for use prior to addition to storage tanks. ITS 5.5.11.a.3 is proposed to be expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil instead of the "clear and bright" test. ASTM D4176-86,"Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Clear and Bright Pass/Fail Procedures)," verifies that the new fuel oil has a clear and bright appearance with proper color. The "clear and bright" test is only applicable to fuel oils that meet the ASTM D4176 color rating requirements (i.e., an ASTM D1500, "Test Method for ASTM Color of Petroleum Products (ASTM Color Scale)," color rating of five or less). The “clear and bright” test is a qualitative test for determining free water and particulate contamination in distillate fuels and is, therefore, subject to human interpretation. For example, if an attempt is made to use the qualitative "clear and bright" test with darker colored fuels (e.g., for high sulfur fuel oil that has been dyed in accordance with EPA mandated requirements), the presence of free water or particulate could be obscured and missed by the viewer. Therefore, ITS 5.5.11.a.3 has been expanded to allow a water and sediment content test. The water and sediment content test is a quantitative test using centrifuge methods. In ASTM D975-90, ASTM D1796, “Standard Method for Water and Sediment in Fuel Oils by the Centrifuge Method (Laboratory Procedure),” is an acceptable standard for the water and sediment content test. In addition, the use of ASTM D1796-83 was endorsed by the NRC in Amendment No. 101 for the Wolf Creek Generating Station. ASTM D1796-83 is the same ASTM Standard used to verify the water and sediment content is within limits within 31 days following sampling and addition to the storage tanks as required by CTS 6.16.b. Therefore, since ASTM D1796 is currently used to verify the acceptability of new fuel oil for use after addition to the storage tanks, the use of these quantitative methods (i.e., water and sediment content) in lieu of ASTM D4176 (i.e., "clear and bright" test) does not introduce a different method for determining the acceptability of new fuel oil. This change is designated as less restrictive because Surveillance acceptance criteria required in the CTS will have alternative acceptance criteria allowed in ITS.

L03 ~~*(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 6.8.4.h specifies the limit for peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig. Also, CTS 6.8.4.h specifies, in part, that bypass leakage paths to the auxiliary building from the isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig). CTS 6.8.4.h.d.2 requires penetrations not individually testable to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test. ITS 5.5.14.a specifies, in part, that bypass leakage paths to the auxiliary building from the isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined~~

Not used.

SII

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SII

~~leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (12.46 psig). ITS 5.5.14.b specifies the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is 11.33 psig and the containment design pressure is 12.0 psig. ITS 5.5.14.d.4.b requires penetrations not individually testable to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (11.33 psig) during each Type A test. This changes the CTS by reducing the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a to 11.33 psig and specifying the containment design pressure is 12.0 psig.~~

~~The purpose of ITS 5.5.14 is to ensure the appropriate limits are specified for the Containment Leakage Rate Testing Program. This change is acceptable because the acceptable limits continue to ensure the containment leakage is within the value assumed in the accident analysis as described in the recent application to modify Ice Condenser Technical Specifications (ML13199A281). Currently, SQN is using the containment design pressure value of 12.0 psig as P_a. In the ITS, the value of P_a (11.33 psig) is the calculated peak containment internal pressure for the design basis loss of coolant accident. This is acceptable because the value of P_a (11.33 psig) is the value assumed in the accident analyses. This change is designated as less restrictive because a lower pressure will be used for P_a in the Containment Leakage Rate Testing Program.~~

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

1.18 5.5.1 Offsite Dose Calculation Manual (ODCM)

1.18 a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program~~,~~; and 1

1.18 b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[~~5.6.1~~]~~ and Specification ~~[~~5.6.2~~]~~. 2

6.14.1.1 Licensee initiated changes to the ODCM:

6.14.1.1 a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

6.14.1.1.a 1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

6.14.1.1.b 2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.1.2 b. Shall become effective after the approval of the plant manager, and

6.14.1.3 c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~[Recirculation Spray,~~ Safety Injection, Chemical and Volume Control, ~~gas stripper, and Hydrogen Recombiner]~~. The program shall include the following: 2

Residual Heat Removal, Containment Spray, and RCS Sampling

6.8.4.a a. Preventive maintenance and periodic visual inspection requirements and

6.8.4.b b. Integrated leak test requirements for each system at least once per ~~[~~18~~]~~ months.

The provisions of SR 3.0.2 are applicable.

~~5.5.3~~ ~~[Post Accident Sampling~~

~~-----REVIEWER'S NOTE-----~~
~~This program may be eliminated based on the implementation of WCAP-14986, Rev. 1, "Post Accident Sampling System Requirements: A Technical Basis," and the associated NRC Safety Evaluation dated June 14, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [Plant] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~ 4

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment.~~
~~-----~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3~~ ~~Post Accident Sampling (continued)~~

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

4

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program 4

6.8.4.f This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

6.8.4.f.1) a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

6.8.4.f.2) b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

6.8.4.f.3) c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

6.8.4.f.4) d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I,

6.8.4.f.5) e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

6.8.4.f.6) f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program (continued) 4

6.8.4.f.7) g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

6.8.4.f.7)).1 1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

6.8.4.f.7)).2 2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,

6.8.4.f.8) h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,

6.8.4.f.9) i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

6.8.4.f.10) j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.8.4.f The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.4.l 5.5.~~5~~ 4 Component Cyclic or Transient Limit 4

This program provides controls to track the U FSAR, Section ~~[ ]~~ 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits. 2

~~5.5.6~~ ~~[Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~ 4

~~The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]~~

5.5 Programs and Manuals

4.0.5.c 5.5.~~7~~ 5 **Reactor Coolant Pump Flywheel Inspection Program** 4

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place ~~UT~~ ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (~~MT~~ magnetic particle and/or ~~PT~~ liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals. 3 3

~~-----REVIEWER'S NOTE-----~~
~~The inspection interval and scope for RCP flywheels stated above can be applied to plants that satisfy the requirements in WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination."~~
~~-----~~ 5

4.0.5 5.5.~~8~~ 6 **Inservice Testing Program** 4

4.0.5 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves. The program shall include the following: 6

4.0.5.b a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

4.0.5.c b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

4.0.5.d c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

5.5 Programs and Manuals

4.0.5 5.5.~~8~~ 6 Inservice Testing Program (continued) 4

4.0.5.e d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

6.8.4.k 5.5.~~9~~ 7 Steam Generator (SG) Program 4

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

6.8.4.k.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected~~,~~ or plugged, ~~[or repaired]~~ to confirm that the performance criteria are being met. 2 1 2

6.8.4.k.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.k.b.1 1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), ~~and~~ all anticipated transients included in the design specification~~)~~, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. TSTF-510-A

6.8.4.k.b.2 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

exceed [1 gpm] per SG ~~[, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program~~. 2 TSTF-510-A

6.8.4.k.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.k.c c. Provisions for SG tube plugging ~~[or repair]~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding [40%] of the nominal tube wall thickness shall be plugged ~~[or repaired]~~. TSTF-510-A 2

~~-----REVIEWER'S NOTE-----~~
~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~
~~----------~~ TSTF-510-A 5

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~

~~1. . . .]~~

6.8.4.k.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A~~n assessment of~~ degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations. 2 TSTF-510-A 7 TSTF-510-A

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

~~------REVIEWER'S NOTE------~~
~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~
~~------~~ 5

6.8.4.k.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation ~~replacement~~.

INSERT 1 ~~[2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.]~~

INSERT 2 ~~[2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.]~~

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6.8.4.k.d.2 ~~[~~2. INSERT 3 ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.]~~

6.8.4.k.d.3 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections ~~is less~~). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. TSTF-510-A

6.8.4.k.e e. Provisions for monitoring operational primary to secondary LEAKAGE.

2

INSERT 1

~~After the first refueling outage following SG installation, inspect each steam generator at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.~~

2

INSERT 2

~~After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.~~

-- Reviewer's Note --

~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;~~

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~~b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and~~
~~c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.~~

INSERT 3

After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. 2 TSTF-510-A

~~--- Reviewer's Note ---~~
~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~
~~---~~

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

~~[f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~ 2

~~------------------------------------REVIEWER'S NOTE------------------------------------~~
~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~
~~--~~ 5

~~1. . . .]~~ 2

6.8.4.c 5.5.~~10~~ 8 Secondary Water Chemistry Program 4

6.8.4.c This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 8

6.8.4.c.(i) a. Identification of a sampling schedule for the critical variables and control points for these variables~~,~~;

6.8.4.c.(ii) b. Identification of the procedures used to measure the values of the critical variables~~,~~;

6.8.4.c.(iii) c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~,~~;

6.8.4.c.(iv) d. Procedures for the recording and management of data~~,~~;

6.8.4.c.(v) e. Procedures defining corrective actions for all off control point chemistry conditions~~,~~; and 11

6.8.4.c.(vi) f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.~~11~~ 9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and~~ in accordance with ~~[~~Regulatory Positions C.5.a, C.5.c, C.5.d and C.6.b of Regulatory Guide 1.52, Revision 2, ~~ASME N510-1989,~~ ANSI N510-1975 and ~~AG-1~~ ASTM D3803-1989~~]~~.

INSERT 4

4.6.1.8.e 4.7.7.c.3 4.7.7.f 4.7.8.b.3 4.7.8.e 4.9.12.b.3 4.9.12.e 4.6.1.8.b.1 4.7.7.c.1 4.7.8.b.1 4.9.12.b.1

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a ~~penetration and system bypass < [0.05]%~~ removal efficiency of ≥ 99.95% of dioctyl phthalate (DOP) when tested in accordance with ~~[~~Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ~~ASME N510-1989]~~ ANSI N510-1975 (except for the provisions of Sections 8 and 9) at the system flowrate specified below ~~[~~± 10%~~]~~.

~~ESF Ventilation System~~ ~~Flowrate~~

~~[ ]~~ ~~[ ]~~ INSERT 5

4.6.1.8.f 4.7.7.c.3 4.7.7.g 4.7.8.b.3 4.7.8.f 4.9.12.b.3 4.9.12.f

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a ~~penetration and system bypass < [0.05]%~~ removal efficiency of ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with ~~[~~Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ~~ASME N510-1989]~~ ANSI N510-1975 (except for the provisions of Sections 8 and 9) at the system flowrate specified below ~~[~~± 10%~~]~~.

~~ESF Ventilation System~~ ~~Flowrate~~

~~[ ]~~ ~~[ ]~~ INSERT 6

4.6.1.8.b.2 4.6.1.8.c 4.7.7.c..2 4.7.7.d 4.7.8.b.2 4.7.8.c 4.9.12.b.2 4.9.12.c

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[~~Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2~~]~~, shows the methyl iodide penetration ~~less than the value specified below~~ < 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity ~~specified below~~ of 70%.

ESF Ventilation System	~~Penetration~~	~~RH~~	~~Face Velocity (fps)~~
~~[ ]~~ EGTS ABGTS CREVS	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~

Air Cleanup Subsystem

CSS-041

~~-----REVIEWER'S NOTE-----~~
~~The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.~~

3 **INSERT 4**

4.6.1.8.b
4.6.1.8.e
4.6.1.8.f
4.7.7.f
4.7.7.g
4.7.8.e
4.7.8.f

The test described in Specification 5.5.9.a and 5.5.9.b shall be performed once per 18 months; after any structural maintenance on the high efficiency particulate air (HEPA) filter bank or charcoal adsorber bank housing; following painting, fire, or chemical release in any ventilation zone communicating with the system; and after each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank.

4.6.1.8.c
4.6.1.8.b
4.7.7.c
4.7.7.d
4.7.8.b
4.7.8.c
4.9.12.b
4.9.12.c

The test described in Specification 5.5.9.c shall be performed once per 18 months or after 720 hours of filter operation; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

4.6.1.8.d
4.7.7.e
4.7.8.d
4.9.12.d

The test described in Specification 5.5.9.d and 5.5.9.e shall be performed once per 18 months.

3 **INSERT 5**

Air Cleanup Subsystem

ESF Ventilation System	Flow Rate (cfm)
Emergency Gas Treatment System (EGTS)	4000
Auxiliary Building Gas Treatment System (ABGTS)	9000
Control Room Emergency Ventilation System (CREVS)	4000

CSS-041

3 **INSERT 6**

ESF Ventilation System	Flow Rate (cfm)
EGTS	4000
ABGTS	9000
CREVS	4000

Air Cleanup Subsystem

5.5 Programs and Manuals

5.5.~~11~~ 9 Ventilation Filter Testing Program (continued) 4

~~ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~

~~When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:~~

~~Safety factor ≥ 2 for systems with or without humidity control.~~

~~Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst-case design-basis conditions.~~

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~ 5

4.6.1.8.d.1
4.7.7.e.1
4.7.8.b.3
4.7.8.d.1
4.9.12.b.3
4.9.12.d.1
4.7.7.c.3
SII

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, ~~the prefilters,~~ and the charcoal adsorbers is less than the value specified below when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-~~1989]~~ 1975 at the system flowrate specified below ~~[~~± 10%~~]~~. 3 2

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~	~~[ ]~~

INSERT 7 3

4.7.8.d.4
4.9.12.d.3

~~[~~ e. Demonstrate that the heaters for ~~each of~~ the ESF system~~s~~ dissipate the value specified below ~~[~~± 10%~~]~~ when tested in accordance with ~~[ASME N510-1989]~~ ANSI N510-1975.

ESF Ventilation System	Wattage ~~]~~
~~[ ]~~ Auxiliary Building Gas Treatment System	~~[ ]~~ 32 kW

2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

CTS 5.5

③ **INSERT 7**

ESF Ventilation System	Combined Delta P (inches water gauge)	Flowrate (cfm)
Air Cleanup Subsystem		CSS-041
EGTS	5	4000
ABGTS	3	9000
CREVS	3	4000

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5.5 Programs and Manuals

3.11.1.4
3.11.2.5
3.11.2.6

5.5.~~12~~ 10 Explosive Gas and Storage Tank Radioactivity Monitoring Program 4

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas ~~storage~~ decay tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in ~~unprotected~~ the Condensate Storage Tank, Steam Generator Layup Tank, and outdoor temporary liquid storage tanks]. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]~~. 2 9

The program shall include:

4.11.2.5 a. The limits for concentrations of hydrogen and oxygen in the [Waste Gas Holdup System] and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion)~~,~~; 2 11

4.11.2.6 b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas ~~storage~~ decay tank ~~and fed into the offgas treatment system~~] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents~~],~~; and 2 2 11

4.11.1.4 c. A surveillance program to ensure that , Condensate Storage Tank, and Steam Generator Layup Tank the quantity of radioactivity contained in all outdoor temporary liquid radwaste storage tanks ~~that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System]~~ is less than the amount that would result in concentrations ~~less than~~ exceeding the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

6.16 5.5.~~13~~ 11 **Diesel Fuel Oil Testing Program** 4

6.16 A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

6.16.a a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

6.16.a.1 1. An API gravity or an absolute specific gravity within limits,

6.16.a.2 2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

6.16.a.3 3. A clear and bright appearance with proper color or a water and sediment content within limits.

6.16.b b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.16.c c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j 5.5.~~14~~ 12 **Technical Specifications (TS) Bases Control Program** 4

6.8.4.j This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.a a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.b b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.b.1 1. A change in the TS incorporated in the license or

6.8.4.j.b.2 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

6.8.4.j.d 5.5.~~14~~ 12 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.d c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 3

6.8.4.j.d d. Proposed changes that meet the criteria of Specification 5.5.~~14~~ 12 b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). 4

DOC M02 5.5.~~15~~ 13 Safety Function Determination Program (SFDP) 4

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected~~,~~ ;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists~~,~~ ;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities~~,~~ ; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable~~,~~ ; and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable~~,~~ ; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable~~,~~ ; or

11

5.5 Programs and Manuals

DOC M02 5.5.~~15~~ 13 Safety Function Determination Program (SFDP) (continued) 4

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5. ~~16~~ 14 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests.~~ 10

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~

~~b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].~~

~~d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

5.5 Programs and Manuals

6.8.4.h 5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4 10

~~[OPTION B]~~

6.8.4.h a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~ INSERT 8

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~ 3

~~[3. . . .]~~

6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident, ~~P_a,~~ is ~~[45 psig]~~ 11.33 less than the ~~. The containment design pressure is~~ of ~~[50~~ 12 psig~~]~~. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig. 2 12

SII

6.8.4.h c. The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~[ ]~~ 0.25 % of containment air weight per day. 2

6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ~~<~~ ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests. 3 3

6.8.4.h.b 2. Air lock testing acceptance criteria are:

6.8.4.h.b.1 a) Overall air lock leakage rate is ≤ ~~[~~0.05 L_a~~]~~ when tested at ≥ P_a. 2

6.8.4.h.b.2 b) For each door, leakage rate is ≤ ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~≥ ~~10~~ 6 psig~~]~~ for at least two minutes. 3

INSERT 9 3

3 **INSERT 8**

6.8.4.h 1. Bypass leakage paths to the auxiliary building leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (~~12.46~~ 13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days. SII

3 **INSERT 9**

6.8.4.h.c 3. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate ≤ to 0.05 L_a.

6.8.4.h.d 4. Bypass leakage paths to the auxiliary building acceptance criteria are:

6.8.4.h.d.1 a) The combined bypass leakage rate to the auxiliary building shall be ≤ 0.25 L_a by applicable Type B and C tests.

6.8.4.h.d.2 b) Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (~~11.33~~ 12 psig) during each Type A test. SII

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5.5 Programs and Manuals

6.8.4.h 5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h ~~e~~. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h ~~f~~. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

10

~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

~~d. Leakage rate acceptance criteria are:~~

5.5 Programs and Manuals

5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and [< 0.75 L_a for Option A Type A tests] [≤ 0.75 L_a for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~ 10

~~b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].~~

~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

DOC M03 5.5.~~17~~ 15 Battery Monitoring and Maintenance Program 4

~~REVIEWER'S NOTE~~
~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~ 5

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. Battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.

5.5 Programs and Manuals

DOC M01 5.5.~~17~~ 15 Battery Monitoring and Maintenance Program (continued) 4

3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."

4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V; 2

Table 4.8.2 Float Voltage

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V; 2

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

6.17 5.5.~~18~~ 16 Control Room Envelope (CRE) Habitability Program 4

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation ~~Filtration~~ System (CREV~~F~~S), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body]~~ ~~[~~5 rem total 3 2

5.5 Programs and Manuals

6.17 5.5.~~18~~ 16 Control Room Envelope (CRE) Habitability Program (continued) 4

effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements: 2

6.17.a a. The definition of the CRE and the CRE boundary.

6.17.b b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

6.17.c c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~ 2

~~1. ;and]~~ 2

6.17.d d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRE~~F~~ V S, operating at the flow rate required by the VFTP, at a Frequency of ~~[~~18~~]~~ months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the ~~[~~18~~]~~ month assessment of the CRE boundary. 3 2 2

6.17.e e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

6.17.f f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals

~~5.5.19~~ ~~[Setpoint Control Program~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation;"~~
~~2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions;"~~
~~3. LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation;"~~
~~4. LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation;"~~
~~5. LCO 3.3.7, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation;"~~
~~6. LCO 3.3.8, "Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation;" and~~
~~7. LCO 3.3.9, "Boron Dilution Protection System (BDPS)."~~

~~b. The program shall require the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.~~

~~-----------------------------------Reviewer's Note-----------------------------------~~
~~List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.~~
~~--~~

-

~~1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]~~

~~c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.~~

4

5.5 Programs and Manuals

~~5.5.19 Setpoint Control Program (continued)~~

~~d. ---------------------------------REVIEWER'S NOTE-----------------------------------~~
~~A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Trip System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~
~~---~~

~~The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.~~

~~1 The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.~~

4

5.5 Programs and Manuals

~~5.5.19 Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~ 4

DOC M04 5.5.~~20~~ 17 ~~[~~ Surveillance Frequency Control Program 4 2

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~ 2

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

1.18 5.5.1 Offsite Dose Calculation Manual (ODCM)

1.18 a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program~~,~~; and 1

1.18 b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[~~5.6.1~~]~~ and Specification ~~[~~5.6.2~~]~~. 2

6.14.1.1 Licensee initiated changes to the ODCM:

6.14.1.1 a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

6.14.1.1.a 1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

6.14.1.1.b 2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.1.2 b. Shall become effective after the approval of the plant manager, and

6.14.1.3 c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~[Recirculation Spray,~~ Safety Injection, Chemical and Volume Control, ~~gas stripper, and Hydrogen Recombiner]~~ Residual Heat Removal, Containment Spray, and RCS Sampling. The program shall include the following: 2

6.8.4.a a. Preventive maintenance and periodic visual inspection requirements and

6.8.4.b b. Integrated leak test requirements for each system at least once per ~~[~~18~~]~~ months.

The provisions of SR 3.0.2 are applicable.

~~5.5.3~~ ~~[Post Accident Sampling~~ 4

~~-----REVIEWER'S NOTE-----~~

~~This program may be eliminated based on the implementation of WCAP-14986, Rev. 1, "Post Accident Sampling System Requirements: A Technical Basis," and the associated NRC Safety Evaluation dated June 14, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [Plant] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment.~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3 Post Accident Sampling (continued)~~

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~ 4

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program 4

6.8.4.f This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

6.8.4.f.1) a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

6.8.4.f.2) b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

6.8.4.f.3) c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

6.8.4.f.4) d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I,

6.8.4.f.5) e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

6.8.4.f.6) f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program (continued) 4

6.8.4.f.7) g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

6.8.4.f.7)).1 1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

6.8.4.f.7)).2 2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,

6.8.4.f.8) h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,

6.8.4.f.9) i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

6.8.4.f.10) j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.8.4.f The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.4.l 5.5.~~5~~ 4 Component Cyclic or Transient Limit 4

This program provides controls to track the U FSAR, Section ~~[ ]~~ 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits. 2

~~5.5.6~~ ~~[Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~ 4

~~The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]~~

5.5 Programs and Manuals

4.0.5.c 5.5.~~7~~ 5 **Reactor Coolant Pump Flywheel Inspection Program** 4

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place ~~UT~~ ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (~~MT~~ magnetic particle and/or ~~PT~~ liquid penetrant) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals. 3 3

~~-----REVIEWER'S NOTE-----~~
~~The inspection interval and scope for RCP flywheels stated above can be applied to plants that satisfy the requirements in WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination."~~
~~----------~~ 5

4.0.5 5.5.~~8~~ 6 **Inservice Testing Program** 4

4.0.5 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves. The program shall include the following: 6

4.0.5.b a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

4.0.5.c b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

4.0.5.d c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

5.5 Programs and Manuals

4.0.5 5.5.~~8~~ 6 Inservice Testing Program (continued) 4

4.0.5.e d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

6.8.4.k 5.5.~~9~~ 7 Steam Generator (SG) Program 4

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

6.8.4.k.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected~~;~~ or plugged, ~~[or repaired]~~ to confirm that the performance criteria are being met. 2 1 2

6.8.4.k.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.k.b.1 1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), ~~and~~ all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. TSTF-510-A

6.8.4.k.b.2 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

exceed [1 gpm] per SG ~~[, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program~~.] 2 TSTF-510-A

6.8.4.k.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.k.c c. Provisions for SG tube ~~repair~~ plugging [or] criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding [40%] of the nominal tube wall thickness shall be plugged ~~[or repaired]~~. TSTF-510-A 2

~~-----REVIEWER'S NOTE-----~~
~~Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.~~
~~----------~~ TSTF-510-A 5

~~[The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:~~

~~1. . . .]~~

6.8.4.k.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging [or] ~~repair~~] criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A~~n assessment of~~ degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations. 2 TSTF-510-A 7 TSTF-510-A

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

~~-----REVIEWER'S NOTE-----
Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.
----------~~ 5

6.8.4.k.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation ~~replacement~~.

INSERT 1

~~[2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.]~~

INSERT 2

~~[2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.]~~

TSTF-510-A 2

6.8.4.k.d.2 ~~[~~2. INSERT 3 ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.]~~

6.8.4.k.d.3 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections ~~is less~~). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. TSTF-510-A

6.8.4.k.e e. Provisions for monitoring operational primary to secondary LEAKAGE.

2

INSERT 1

~~After the first refueling outage following SG installation, inspect each steam generator at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.~~

2

INSERT 2

~~After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.~~

TSTF-510-A

-- Reviewer's Note --

~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;~~

~~b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and~~
~~c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.~~

INSERT 3

After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. 2 TSTF-510-A

~~--- Reviewer's Note ---~~
~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~
~~---~~

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

5.5 Programs and Manuals

5.5.~~9~~ 7 Steam Generator (SG) Program (continued) 4

~~[f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~ 2

~~-----REVIEWER'S NOTE-----~~
~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~
~~----------~~ 5

~~1. . . .]~~ 2

6.8.4.c 5.5.~~10~~ 8 Secondary Water Chemistry Program 4

6.8.4.c This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 8

6.8.4.c.(i) a. Identification of a sampling schedule for the critical variables and control points for these variables~~,~~ ;

6.8.4.c.(ii) b. Identification of the procedures used to measure the values of the critical variables~~,~~ ;

6.8.4.c.(iii) c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~,~~ ; 11

6.8.4.c.(iv) d. Procedures for the recording and management of data~~,~~ ;

6.8.4.c.(v) e. Procedures defining corrective actions for all off control point chemistry conditions~~,~~ ; and

6.8.4.c.(vi) f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.~~11~~ 9 Ventilation Filter Testing Program (VFTP)

Regulatory Positions C.5.a, C.5.c, C.5.d and C.6.b of

4 3

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and~~ in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, ~~ASME N510-1989,~~ ANSI N510-1975 and ~~AG-1~~ ASTM D3803-1989~~]~~.

2

INSERT 4

removal efficiency of ≥ 99.95% of dioctyl phthalate (DOP)

3

4.6.1.8.e
4.7.7.c.3
4.7.7.f
4.7.8.b.3
4.7.8.e
4.9.12.b.3
4.9.12.e
4.6.1.8.b.1
4.7.7.c.1
4.7.8.b.1
4.9.12.b.1

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a ~~penetration and system bypass < [0.05]%~~ removal efficiency of ≥ 99.95% of dioctyl phthalate (DOP) when tested in accordance with ~~[~~Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ~~ASME N510-1989]~~ ANSI N510-1975 (except for the provisions of Sections 8 and 9) at the system flowrate specified below ~~[~~± 10%~~]~~.

2 3 2

~~ESF Ventilation System~~ ~~Flowrate~~

~~[ ]~~ ~~[ ]~~

INSERT 5

2

removal efficiency of ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas

3

4.6.1.8.f
4.7.7.c.3
4.7.7.g
4.7.8.b.3
4.7.8.f
4.9.12.b.3
4.9.12.f

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a ~~penetration and system bypass < [0.05]%~~ removal efficiency of ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with ~~[~~Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ~~ASME N510-1989]~~ ANSI N510-1975 (except for the provisions of Sections 8 and 9) at the system flowrate specified below ~~[~~± 10%~~]~~.

2 2 3

~~ESF Ventilation System~~ ~~Flowrate~~

~~[ ]~~ ~~[ ]~~

INSERT 6

3

Regulatory Position C.6.b of

3

4.6.1.8.b.2
4.6.1.8.c
4.7.7.c..2
4.7.7.d
4.7.8.b.2
4.7.8.c
4.9.12.b.2
4.9.12.c

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[~~Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2~~]~~, shows the methyl iodide penetration ~~less than the value specified below~~ < 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity ~~specified below~~ of 70%.

2 3 3

ESF Ventilation System ~~Penetration~~ ~~RH~~ ~~Face Velocity (fps)~~

~~[ ]~~ ~~[See Reviewer's Note]~~ ~~[See Reviewer's Note]~~ ~~[See Reviewer's Note]~~

EGTS
ABGTS
CREVS

CSS-041

Air Cleanup Subsystem

2 3

~~REVIEWER'S NOTE~~

~~The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.~~

5

3 INSERT 4

4.6.1.8.b
4.6.1.8.e
4.6.1.8.f
4.7.7.f
4.7.7.g
4.7.8.e
4.7.8.f

The test described in Specification 5.5.9.a and 5.5.9.b shall be performed once per 18 months; after any structural maintenance on the high efficiency particulate air (HEPA) filter bank or charcoal adsorber bank housing; following painting, fire, or chemical release in any ventilation zone communicating with the system; and after each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank.

4.6.1.8.c
4.6.1.8.b
4.7.7.c
4.7.7.d
4.7.8.b
4.7.8.c
4.9.12.b
4.9.12.c

The test described in Specification 5.5.9.c shall be performed once per 18 months or after 720 hours of filter operation; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

4.6.1.8.d
4.7.7.e
4.7.8.d
4.9.12.d

The test described in Specification 5.5.9.d and 5.5.9.e shall be performed once per 18 months.

3 INSERT 5

Air Cleanup Subsystem

CSS-041

ESF Ventilation System	Flow Rate (cfm)
Emergency Gas Treatment System (EGTS)	4000
Auxiliary Building Gas Treatment System (ABGTS)	9000
Control Room Emergency Ventilation System (CREVS)	4000

3 INSERT 6

ESF Ventilation System	Flow Rate (cfm)
EGTS Air Cleanup Subsystem	4000
ABGTS	9000
CREVS	4000

5.5 Programs and Manuals

5.5.~~11~~ 9 Ventilation Filter Testing Program (continued) 4

~~ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~

~~When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:~~

~~Safety factor ≥ 2 for systems with or without humidity control.~~

~~Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst-case design-basis conditions.~~

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~

5

4.6.1.8.d.1
4.7.7.e.1
4.7.8.b.3
4.7.8.d.1
4.9.12.b.3
4.9.12.d.1
4.7.7.c.3

SII

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, ~~the prefilters,~~ and the charcoal adsorbers is less than the value specified below when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-~~1989]~~ 1975 at the system flowrate specified below ~~[~~± 10%~~]~~. 3 2

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~	~~[ ]~~

INSERT 7

3

4.7.8.d.4
4.9.12.d.3

~~[~~ e. Demonstrate that the heaters for ~~each of~~ the ESF system~~s~~ dissipate the value specified below ~~[~~± 10%~~]~~ when tested in accordance with ~~[ASME N510-1989]~~ ANSI N510-1975.

ESF Ventilation System	Wattage ~~]~~
~~[ ]~~ Auxiliary Building Gas Treatment System	~~[ ]~~ 32 kW

2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

3 **INSERT 7**

CSS-041 Air Cleanup Subsystem

ESF Ventilation System	Combined Delta P (inches water gauge)	Flowrate (cfm)
EGTS	5	4000
ABGTS	3	9000
CREVS	3	4000

Enclosure 2, Volume 16, Rev. 1, Page 191 of 270

5.5 Programs and Manuals

3.11.1.4
3.11.2.5
3.11.2.6

5.5.~~12~~ 10 Explosive Gas and Storage Tank Radioactivity Monitoring Program 4

This program provides controls for potentially explosive gas mixtures contained in the ~~[~~Waste Gas Holdup System~~]~~, ~~[~~the quantity of radioactivity contained in gas decay ~~storage~~ tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in the Condensate Storage Tank, Steam Generator Layup Tank, and ~~unprotected~~ outdoor temporary liquid storage tanks~~]~~. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]~~. 2 9

The program shall include:

4.11.2.5

a. The limits for concentrations of hydrogen and oxygen in the ~~[~~Waste Gas Holdup System~~]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion)~~,~~ ; 2 11

4.11.2.6

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each gas decay ~~storage~~ tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~, ; and 2 2 11

4.11.1.4

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor temporary liquid radwaste storage tanks , Condensate Storage Tank, and Steam Generator Layup Tank ~~that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System]~~ is less than the amount that would result in concentrations ~~less than~~ exceeding the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

6.16 5.5.~~13~~ 11 Diesel Fuel Oil Testing Program 4

6.16 A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

6.16.a a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

6.16.a.1 1. An API gravity or an absolute specific gravity within limits,

6.16.a.2 2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

6.16.a.3 3. A clear and bright appearance with proper color or a water and sediment content within limits.

6.16.b b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.16.c c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j 5.5.~~14~~ 12 Technical Specifications (TS) Bases Control Program 4

6.8.4.j This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.a a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.b b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.b.1 1. A change in the TS incorporated in the license or

6.8.4.j.b.2 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

6.8.4.j.d 5.5.~~14~~ 12 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.d c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 3

6.8.4.j.d d. Proposed changes that meet the criteria of Specification 5.5.~~14~~ 12 b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). 4

DOC M02 5.5.~~15~~ 13 Safety Function Determination Program (SFDP) 4

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected~~,~~ ;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists~~,~~ ;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities~~,~~ ; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable~~,~~ ; and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable~~,~~ ; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable~~,~~ ; or

11

5.5 Programs and Manuals

DOC M02 5.5.~~15~~ 13 Safety Function Determination Program (SFDP) (continued) 4

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5. ~~16~~ 14 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate,~~ $\cancel{L_a}$~~, at~~ $\cancel{P_a}$~~, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is~~ $\cancel{\leq 1.0\ L_a}$~~. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are~~ $\cancel{< 0.60\ L_a}$ ~~for the Type B and C tests and~~ $\cancel{< 0.75\ L_a}$ ~~for Type A tests.~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is~~ $\cancel{\leq [0.05\ L_a]}$ ~~when tested at~~ $\cancel{\geq P_a}$~~.~~

~~b) For each door, leakage rate is~~ $\cancel{\leq [0.01\ L_a]}$ ~~when pressurized to [≥ 10 psig].~~

~~d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

10

5.5 Programs and Manuals

6.8.4.h 5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4 10

~~[OPTION B]~~

6.8.4.h a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~ INSERT 8

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~ 3

6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident~~, P_a,~~ is less than the ~~[45 psig]~~ ~~11.33~~. ~~The~~ containment design pressure ~~is~~ of ~~[50~~ 12 psig~~]~~. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig. 2 12

SII

6.8.4.h c. The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~[ ]~~ 0.25% of containment air weight per day. 2

6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ~~<~~ ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests. 3 3

6.8.4.h.b 2. Air lock testing acceptance criteria are:

6.8.4.h.b.1 a) Overall air lock leakage rate is ≤ ~~[~~0.05 L_a~~]~~ when tested at ≥ P_a.

6.8.4.h.b.2 b) For each door, leakage rate is ≤ ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~≥ ~~10~~ 6 psig~~]~~ for at least two minutes. 2 3

INSERT 9 3

3 INSERT 8

6.8.4.h 1. Bypass leakage paths to the auxiliary building leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (~~12.46~~ 13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

SII

3 INSERT 9

6.8.4.h.c 3. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate ≤ to 0.05 L_a.

6.8.4.h.d 4. Bypass leakage paths to the auxiliary building acceptance criteria are:

6.8.4.h.d.1 a) The combined bypass leakage rate to the auxiliary building shall be ≤ 0.25 L_a by applicable Type B and C tests.

6.8.4.h.d.2 b) Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (~~11.33~~ 12 psig) during each Type A test.

SII

5.5 Programs and Manuals

6.8.4.h 5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h ~~e~~. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h ~~f~~. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

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~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

~~d. Leakage rate acceptance criteria are:~~

5.5 Programs and Manuals

5.5.~~16~~ 14 Containment Leakage Rate Testing Program (continued) 4

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and [< 0.75 L_a for Option A Type A tests] [≤ 0.75 L_a for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~

~~b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].~~

~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~ 10

DOC M03 5.5.~~17~~ 15 Battery Monitoring and Maintenance Program 4

~~REVIEWER'S NOTE~~
~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~ 5

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. Battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.

5.5 Programs and Manuals

DOC M01 5.5.~~17~~ 15 Battery Monitoring and Maintenance Program (continued) 4

3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."

4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V; 2

Table 4.8.2 Float Voltage

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V; 2

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

6.17 5.5.~~18~~ 16 Control Room Envelope (CRE) Habitability Program 4

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency ~~Filtration~~ Ventilation System (CRE~~F~~VS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body]~~ ~~[~~5 rem total 3 2

5.5 Programs and Manuals

6.17 5.5.~~18~~ 16 Control Room Envelope (CRE) Habitability Program (continued) 4

effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements: 2

6.17.a a. The definition of the CRE and the CRE boundary.

6.17.b b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

6.17.c c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~ 2

~~1. ;and]~~ 2

6.17.d d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRE~~F~~VS, operating at the flow rate required by the VFTP, at a Frequency of ~~[~~18~~]~~ months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the ~~[~~18~~]~~ month assessment of the CRE boundary. 3 2 2

6.17.e e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

6.17.f f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals

~~5.5.19~~ ~~[Setpoint Control Program~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation;"~~
~~2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions;"~~
~~3. LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation;"~~
~~4. LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation;"~~
~~5. LCO 3.3.7, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation;"~~
~~6. LCO 3.3.8, "Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation;" and~~
~~7. LCO 3.3.9, "Boron Dilution Protection System (BDPS)."~~

~~b. The program shall require the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.~~

~~------------------------------------Reviewer's Note------------------------------------~~
~~List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.~~
~~---~~

-

~~1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]~~

~~c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.~~

4

5.5 Programs and Manuals

~~5.5.19 Setpoint Control Program (continued)~~

~~d. ------------------------------REVIEWER'S NOTE-----------------------------------~~
~~A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Trip System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~

~~---~~

~~The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.~~

~~1 The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.~~

4

5.5 Programs and Manuals

~~5.5.19 Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~

4

DOC M04 5.5.~~20~~ 17 ~~[~~ Surveillance Frequency Control Program 4 2

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. Typographical/grammatical error corrected.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The bracketed ISTS 5.5.3, "Post Accident Sampling," ISTS 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," and ISTS 5.5.19, "Setpoint Control Program," are not included in the SQN ITS. Subsequent programs in the ITS Section 5.5 have been renumbered, as necessary.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. The Inservice Testing (IST) Program (ISTS 5.5.8) has been modified to state that the IST Program provides control for ASME Code Class 1, 2, and 3 "pumps and valves" in place of the current "components." 10 CFR 50.55a(f) provides the regulatory requirements for an IST Program. It specifies that ASME Code Class 1, 2, and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program. It specifies that ASME Code Class 1, 2, and 3 components are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements as these requirements have been relocated to a plant specific document. Therefore, the components to which the IST Program applies (i.e., pumps and valves) have been added for clarity.

7. Changes made to improve clarity.

8. ISTS 5.5.10 (ITS 5.5.8) provides the requirements for the Secondary Water Chemistry Program. The program in the ISTS includes requirements to provide controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion. ITS 5.5.8 provides controls for monitoring secondary water chemistry only to inhibit SG tube degradation. This change is consistent with the current SQN licensing bases.

9. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.12 (ITS 5.5.10) parts a, b, and c. Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary.

10. SQN complies with Option B of 10 CFR 50, Appendix J. Therefore, the ISTS 5.5.16 Option A and combined Option A and B provisions have been deleted.

11. These punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications NEI 01-03, Section 5.1.3.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

SII

12. ISTS 5.5.16 (ITS 5.5.14) provides the requirements for the Containment Leakage Rate Testing Program. ISTS 5.5.16.b states, "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig]." ISTS 5.5.16.b is revised to state, "The calculated peak containment internal pressure for the design basis loss of coolant accident is less than the containment design pressure of 12 psig. For the Containment Leakage Rate Testing Program, P_a is defined as 12.0 psig." By letter dated February 5, 1996, the NRC issued Amendments 217 and 207 to SQN, Units 1 and 2, respectively, approving the establishment of a "Containment Leakage Rate Testing Program." The program is CTS 6.8.4.h, and states, "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig." Although SQN performs containment leakage rate testing using a P_a of 12.0 psig, the <u>calculated</u> peak containment internal pressure for the design basis loss of coolant accident is a value less than 12.0 psig. Therefore, ISTS 5.5.16.b is revised to align P_a with the value used in CTS 6.8.4.h. The <u>calculated</u> peak containment internal pressure for the design basis loss of coolant accident will be stated in the Applicable Safety Analyses Section of the Bases for ITS Specifications 3.6.4, Containment Pressure, and 3.6.6, Containment Spray.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 6

ITS 5.6, REPORTING REQUIREMENTS

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

ITS A01 ITS 5.6

6.0 ADMINISTRATIVE CONTROLS

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. See ITS 5.5

e. Provisions for monitoring operational primary-to-secondary leakage.

l. Component Cyclic and Transient Limit

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.6 6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

5.6 6.9.1 ~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations,~~ the following reports shall be submitted in accordance with 10 CFR 50.4.

~~STARTUP REPORT~~

~~6.9.1.1 DELETED~~

~~6.9.1.2 DELETED~~

~~6.9.1.3 DELETED~~

~~ANNUAL REPORTS [1/]~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ A02

~~6.9.1.5 DELETED~~

~~1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~ A02

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5.6.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT 1/

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~prior to May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. L01

INSERT 1 M01

~~6.9.1.7 (Relocated to the ODCM.)~~

5.6.2 ~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT 1/

6.9.1.8 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50. A03

~~6.9.1.9 (Relocated to the ODCM or PCP.)~~

5.6.1 Note, 5.6.2 Note

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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M01

INSERT 1

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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ADMINISTRATIVE CONTROLS

~~MONTHLY REACTOR OPERATING REPORT~~

~~6.9.1.10 DELETED.~~

5.6.3 CORE OPERATING LIMITS REPORT

5.6.3.a 6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.

2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

4. Control Bank Insertion Limits for Specification 3/4.1.3.6,

5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,

6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and

7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

5.6.3.b 6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC, specifically those described in the following documents:

~~The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).~~ A04

1. BAW-10180P-A, Revision 1, "NEMO - Nodal Expansion Method Optimized", March 1993

2. BAW-10169P-A, Revision 0, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants", October 1989

3. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs", June 1989

4. ~~BAW-10168P-A, "RSG LOCA – B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants"~~ EMF-2328 (P)(A), "PWR Small Break LOCA Evaluation Model," March 2001

A07

~~SL 2.1.1, "Reactor Core Safety Limits"~~
LCO 3.1.1, SHUTDOWN MARGIN"
~~LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," – Overtemperature ΔT and Overpower ΔT Nominal Trip Setpoint denoted values~~
~~LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"~~
LCO 3.9.1, "Boron Concentration"

A05

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ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

~~5.~~ ~~WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"~~

~~6.~~ ~~WCAP-10266-P-A, "The 1981 Revision of Westinghouse Evaluation Model Using BASH CODE"~~

5 ~~7~~. BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", June 2003

6 ~~8~~. BAW-10186-A, Revision 2, "Extended Burnup Evaluation", June 2003

7 ~~9~~. EMF-2103P-A, Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors", April 2003

A07

INSERT 2

5.6.3.c 6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

M02

5.6.3.d 6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days ~~after cycle start-up (Mode 2)~~ for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector~~.

A06

5.6.4 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

5.6.4.a 6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, "RCS Pressure and Temperature (P/T) Limits"

Specification 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System"

5.6.4.b 6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

5.6.4.c 6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

STEAM GENERATOR TUBE INSPECTION REPORT

5.6.6 6.9.1.16 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, Steam Generator (SG) Program. The report shall include:

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A07 **INSERT 2**

8. BAW-1 0241 P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT," July 2005

9. BAW-10199P-A, Revision 0, "The BWU Critical Heat Flux Correlations," August 1996

10. BAW-10189P-A, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," January 1996

11. BAW-10159P-A, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," August 1990

12. BAW-10231(P)(A), Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004

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5.6.6.a a. The scope of inspections performed on each SG,

5.6.6.b b. Active degradation mechanisms found,

5.6.6.c c. Nondestructive examination techniques utilized for each degradation mechanism,

5.6.6.d d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5.6.6.e e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

5.6.6.f f. Total number and percentage of tubes plugged to date,

5.6.6.g g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

5.6.6.f h. The effective plugging percentage for all plugging in each SG.

~~SPECIAL REPORTS~~

5.6 6.9.2.1 ~~Special~~ reports shall be submitted ~~within the time period specified for each report,~~ in accordance with 10 CFR 50.4.

~~6.9.2.2 This specification has been deleted.~~

~~6.10 RECORD RETENTION (DELETED)~~

ITS

A01

ITS 5.6

TABLE 3.3-10 (Continued)

ACTION STATEMENTS
(Continued)

ACTION 4 - With the number of channels less than the minimum channels required, initiate an alternate method of monitoring containment area radiation within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 30 days, or prepare and submit a special report to the Commission pursuant to Specification 6.9.2.1 within the next 14 days that provides actions taken, cause of the inoperability, and plans and schedule for restoring the channels to OPERABLE status.

See ITS 3.3.3

5.6.5

ACTION 5 - NOTE: Also refer to the applicable action requirements from LCO 3.3.3.5 since it may contain more restrictive actions.

a. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.

b. With the number of channels on one or more steam generators less than the minimum channels required for flow rate and valve position, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.

See ITS 3.3.3

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5.6 6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

5.6 6.9.1 ~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations,~~ the following reports shall be submitted in accordance with 10 CFR 50.4.

~~STARTUP REPORT~~

~~6.9.1.1 DELETED~~

~~6.9.1.2 DELETED~~

~~6.9.1.3 DELETED~~

~~ANNUAL REPORTS[1]~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ A02

~~6.9.1.5 DELETED~~

~~[1]/A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~ A02

ITS

A01

ITS 5.6.

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[1]/

5.6.1 6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~prior to May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. L01

INSERT 1 M01

~~6.9.1.7 (Relocated to the ODCM.)~~

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT

in accordance with 10 CFR 50.36a A03

5.6.2 6.9.1.8 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

~~6.9.1.9 (Relocated to the ODCM or PCP.)~~

5.6.1 Note, 5.6.2 Note [1]/A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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M01

INSERT 1

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 DELETED

5.6.3 CORE OPERATING LIMITS REPORT

5.6.3.a 6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.
2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
4. Control Bank Insertion Limits for Specification 3/4.1.3.6,
5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,
6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and
7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

5.6.3.b 6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC, specifically those described in the following documents:

~~The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).~~ A04

1. BAW-10180P-A, Revision 1, "NEMO - Nodal Expansion Method Optimized," March 1993
2. BAW-10169P-A, Revision 0, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989
3. BAW-10163P-A, Revision 0, “Core Operating Limit Methodology for Westinghouse Designed PWRs,” June 1989
4. EMF-2328(P)(A), “PWR Small Break LOCA Evaluation Model,” March 2001
5. BAW-10227P-A, Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003
6. BAW-10186P-A, Revision 2, “Extended Burnup Evaluation,” June 2003
7. EMF-2103P-A, Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” April 2003

SII

~~SL 2.1.1, “Reactor Core Safety Limits”~~
LCO 3.1.1, SHUTDOWN MARGIN”
~~LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," - Overtemperature ΔT and Overpower ΔT Nominal Trip Setpoint denoted values~~
~~LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB Limits”~~
LCO 3.9.1, “Boron Concentration” A05

ITS A01 ITS 5.6.

CORE OPERATING LIMITS REPORT (continued)

8. BAW-10241P-A, Revision 1, “BHTP DNB Correlation Applied with LYNXT,” July 2005

9. BAW-10199P-A, Revision 0, “The BWU Critical Heat Flux Correlations,” August 1996

10. BAW-10189P-A, “CHF Testing and Analysis of the Mark-BW Fuel Assembly Design,” January 1996

11. BAW-10159P-A, “BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies,” August 1990

12. BAW-10231(P)(A), Revision 1, “COPERNIC Fuel Rod Design Computer Code,” January 2004

5.6.3.c 6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

M02

5.6.3.d 6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days ~~after cycle start-up (Mode 2)~~ for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector~~.

A06

5.6.4 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

5.6.4.a 6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, “RCS Pressure and Temperature (P/T) Limits”

Specification 3.4.12, “Low Temperature Over Pressure Protection (LTOP) System”

5.6.4.b 6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, “Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.”

2. Westinghouse Topical Report WCAP-15321, “Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation.”

3. Westinghouse Topical Report WCAP-15984, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2.”

5.6.4.c 6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

STEAM GENERATOR (SG) TUBE INSPECTION REPORT

5.6.6 6.9.1.16.1 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.k, “Steam Generator (SG) Program.” The report shall include:

5.6.6.a a. The scope of inspections performed on each SG,

5.6.6.b b. Active degradation mechanisms found,

5.6.6.c c. Nondestructive examination techniques utilized for each degradation mechanism,

5.6.6.d d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5.6.6.e e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

5.6.6.f f. Total number and percentage of tubes plugged to date,

5.6.6.g g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

5.6.6.f h. The effective plugging percentage for all plugging in each SG.

~~SPECIAL REPORTS~~

5.6 6.9.2.1 ~~Special~~ reports shall be submitted ~~within the time period specified for each report,~~ in accordance with 10 CFR 50.4.

6.9.2.2 This specification has been deleted.

6.10 RECORD RETENTION (DELETED)

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ITS

A01

ITS 5.6.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS
(Continued)

ACTION 4 - With the number of channels less than the minimum channels required, initiate an alternate method of monitoring containment area radiation within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 30 days, or prepare and submit a special report to the Commission pursuant to Specification 6.9.2.1 within 14 days that provides actions taken, cause of the inoperability, and plans and schedule for restoring the channels to OPERABLE status.

See ITS 3.3.3

5.6.5

ACTION 5 - NOTE: Also refer to the applicable action requirements from LCO 3.3.3.5 since it may contain more restrictive actions.

a. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.

b. With the number of channels on one or more steam generators less than the minimum channels required for flow rate and valve position, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.

See ITS 3.3.3.

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DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 6.9.1.4 states that, annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year following initial criticality. ITS 5.6 does not include the requirement for annual reports. This changes the CTS by not including the requirements.

The purpose of CTS 6.9.1.4 is to specify submittal dates of annual reports for associated activities. This change is acceptable because no activities are associated with the current Specification. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.9.1.8 requires the Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year to be submitted prior to May 1 of each year. ITS 5.6.2 requires this report, the Radioactive Effluent Release Report, covering the operation of the unit in the previous year to be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. This changes the CTS by explicitly stating the report shall be submitted in accordance with 10 CFR 50.36a.

The purpose to CTS 6.9.1.8 is to provide the requirements associated with the Radioactive Effluent Release Report. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," also provides requirements for submission of a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months. 10 CFR 50.36a also states that the time between submissions of the reports must be no longer than 12 months. This change is acceptable because the CTS reporting requirements have not changed, ITS explicitly states that the reporting requirement of "prior to May 1 of each year," is also in accordance with 10 CFR 50.36a. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.9.1.14.a requires, in part, the COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). ITS 5.6.3 b Reviewers Note states, licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). This changes the CTS by not including the requirement of referencing Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). SQN has received prior approval by the NRC to include reference Topical Reports used to prepare the COLR (i.e., report number, title revision, date, and any supplements) in the Specification.

This change is acceptable because the Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements) have been included in the Specification. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 6.9.1.14 contains a list of the core operating limits established and documented in the COLR. ITS 5.6.~~5~~.a includes additional core operating limits established and documented in the COLR. These are ~~Reactor Core Safety Limits;~~ SHUTDOWN MARGIN; ~~Reactor Trip System (RTS) Instrumentation, Overtemperature ΔT and Overpower ΔT Nominal Trip Setpoint denoted values; RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits;~~ and Boron Concentration. These limits had previously been addressed in other parts of the CTS, but are being moved to the COLR in the ITS, and because of this are listed in ITS 5.6.~~5~~.a. This changes the CTS by adding core operating limits established and documented in the COLR because they are being moved there as part of changes to other parts of the CTS. Technical aspects of the changes are addressed in the Discussion of Changes for the respective individual ITS Specifications. 3 SII

This change is acceptable because it administratively documents changes made to other parts of the CTS and the COLR. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS 6.9.1.14.c requires, in part the CORE OPERATING LIMITS REPORT (COLR) to be provided to the NRC document control desk with copies to the Regional Administrator and Resident Inspector. ITS 5.6.3.d requires the COLR to be provided to the NRC. This changes the CTS by removing the specifics regarding distribution of the report to the NRC.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. This change is designated as administrative because it does not result in technical changes to the CTS.

A07 SQN Unit 1 CTS 6.9.1.14.a requires, in part that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, to be listed. TVA has received approval to change the list of approved documents used to determine the core operating limits. This changes the CTS by revising the list of approved documents to those approved in License Amendment 331 before it has been implemented at SQN Unit 1.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

This change is acceptable because this change was approved by License Amendment 331/324 [Unit 1/Unit 2] in September of 2012 by letter titled, "Sequoyah Nuclear Plant, Units 1 and 2 Issuance of Amendments to Revise the Technical Specification to allow use of Areva Advanced W17 High Thermal Performance Fuel (TS-SQN-2011-07) (TAC NOS. ME6538 and ME6539)" (ADAMS Accession No. ML12249A394). This amendment was effective as of its date of issuance, to be implemented on Unit 1 prior to startup from Unit 1 fall 2013 refueling outage and on Unit 2 prior to startup from Unit 2 fall 2012 refueling outage. SQN Unit 2 License Amendment 324 has been implemented on Unit 2 and is reflected in this license amendment request. Because the implementation of SQN Unit 1 License Amendment 331 is after the submittal of the SQN ITS conversion license amendment request, the values approved in License Amendment 331 are shown as being inserted. This change is designated as administrative because it does not result in technical changes to the CTS approved by the NRC.

MORE RESTRICTIVE CHANGES

M01 The second paragraph of ITS 5.6.1 includes details required to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1 does not contain this level of detail. This changes the CTS by requiring additional detail to be included in the Annual Radiological Environmental Operating Report.

The purpose of the second paragraph of ITS 5.6.1 is to specify details to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

M02 CTS 6.9.1.14.c states, in part, that the CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle. ITS 5.6.3.d states, in part, that the COLR shall be provided within 30 days of issuance for each reload cycle to the NRC. This changes the CTS by eliminating the allowance to wait until entering MODE 2 for the 30 day period to begin before requiring the COLR to be submitted to the NRC.

The purpose to CTS 6.9.1.14.c is to provide guidance on when the COLR is required to be submitted to the NRC. ITS 5.6.3.d provides similar guidance but requires the COLR to be submitted in less time than allowed by CTS. This change is acceptable because the ITS requirement for submission of the COLR continues to allow adequate time to process the submittal and is within the CTS requirements. This change is designated as more restrictive because less time is allowed in ITS to submit the COLR than is allowed in CTS.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 L01 *(Category 1 –Relaxation of LCO Requirements)* CTS 6.9.1.6 requires the Annual Radiological Environmental Operating Report to be submitted prior to May 1 of each year. ITS 5.6.1 requires the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. This changes the CTS by allowing additional time to submit this report each year.

The purpose of the due date for submitting the Annual Radiological Environmental Operating Report is to ensure that the report is provided in a reasonable period of time to the NRC for review. This change is acceptable because the report is still required to be provided to the NRC on or before May 15 and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between May 1 and May 15. Additionally, there is no requirement for the NRC to approve the report. This change is designated as less restrictive because it allows more time to prepare and submit the report to the NRC.

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

5.0 ADMINISTRATIVE CONTROLS

6.9

5.6 Reporting Requirements

6.9.1 The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.6

5.6.1 Annual Radiological Environmental Operating Report

6.9.1 Note

---------------------------------------~~REVIEWER'S~~ NOTE---------------------------------------
~~[~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~]~~ 1

6.9.1.6
DOC L01

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[~~in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979~~]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 1

6.9.1.8

5.6.2 Radioactive Effluent Release Report

6.9.1 Note

---------------------------------------~~REVIEWER'S~~ NOTE---------------------------------------
~~[~~A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. ~~]~~ 1

6.9.1.8

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

6.9.1.14 5.6.3 CORE OPERATING LIMITS REPORT

6.9.1.14 a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~[The individual specifications that address core operating limits must be referenced here.]~~ INSERT 1 1

6.9.1.14.a b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~-----REVIEWER'S NOTE-----~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~----------~~ 3

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ INSERT 2 1

6.9.1.14.b c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.14.c d. The COLR, including any midcycle revisions or supplements, shall be provided ~~upon~~ within 30 days of issuance for each reload cycle to the NRC. 2

6.9.1.15 5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT

6.9.1.15 a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ INSERT 3 1

ITS 5.6

1 **INSERT 1**

SII

~~1. SL 2.1.1, "Reactor Core Safety Limits";~~

1 ~~2.~~ LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";

2 ~~3.~~ LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";

3 ~~4.~~ LCO 3.1.5, "Shutdown Bank Insertion Limits";

4 ~~5.~~ LCO 3.1.6, "Control Bank Insertion Limits";

5 ~~6.~~ LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(X, Y, Z)$)";

6 ~~7.~~ LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$)";

7 ~~8.~~ LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";

8 ~~9.~~ LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," $f_1(\Delta I)$ limits for ~~;~~ Overtemperature ΔT and $f_2(\Delta I)$ limits for Overpower ΔT Nominal Trip Setpoints ~~denoted values~~; and

~~10. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and~~

9 ~~11.~~ LCO 3.9.1, "Boron Concentration."

1 INSERT 2

1. BAW-10180P-A, Revision 1, "NEMO - Nodal Expansion Method Optimized," March 1993
2. BAW-10169P-A, Revision 0, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989
3. BAW-10163P-A, Revision 0,"Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989
4. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model," March 2001
5. BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003
6. BAW-10186-A, Revision 2, "Extended Burnup Evaluation," June 2003
7. EMF-2103P-A, Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003
8. BAW-10241P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT," July 2005
9. BAW-10199P-A, Revision 0, "The BWU Critical Heat Flux Correlations," August 1996
10. BAW-10189P-A, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," January 1996
11. BAW-10159P-A, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," August 1990 and
12. BAW-10231(P)(A), Revision 1, "COPERNIC Fuel Rod Design Computer Code" January 2004.

1 INSERT 3

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
2. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System"; and
3. LCO 3.5.2, "ECCS - Operating".

5.6 Reporting Requirements

5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

6.9.1.15.a b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~-----REVIEWER'S NOTE-----~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~----------~~ 3

~~[Identify the NRC staff approval document by date.]~~ INSERT 4 1

within 30 days of 2

6.9.1.15.b c. The PTLR shall be provided to the NRC ~~upon~~ issuance for each reactor vessel fluence period and for any revision or supplement thereto.

~~-----REVIEWER'S NOTE-----~~
~~The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~

~~2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~

~~3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~ 3

~~4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~

~~5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~

~~6. LTOP arming temperature limit development methodology.~~

1 **INSERT 4**

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves";

2. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation"; and

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

5.6 Reporting Requirements

5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

~~7. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

~~8. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~ 3

3.3.10 ACTION 4 5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or ~~F~~ I of LCO 3.3.~~[~~3~~]~~, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. 5 1

~~5.6.6 [Tendon Surveillance Report~~

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ 4

6.9.1.16 5.6.~~7~~ 6 Steam Generator Tube Inspection Report 4

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.~~9~~ 7, "Steam Generator (SG) Program." The report shall include: 6

6.9.1.16.a a. The scope of inspections performed on each SG~~,~~ ;

6.9.1.16.b b. Active degradation mechanisms found~~,~~ ; 8

6.9.1.16.c c. Nondestructive examination techniques utilized for each degradation mechanism~~,~~ ;

5.6 Reporting Requirements

5.6.~~7~~ 6 Steam Generator Tube Inspection Report (continued) 4

6.9.1.16.d d. Location, orientation (if linear), and measured sizes (if available) of service induced indications~~,~~ ; 8

6.9.1.16.e e. Number of tubes plugged ~~[or repaired]~~ during the inspection outage for each active degradation mechanism~~,~~ ;

6.9.1.16.f f. Total number and percentage of tubes plugged ~~[or repaired]~~ to date~~,~~ ;

6.9.1.16.g g. The results of condition monitoring, including the results of tube pulls and in-situ testing~~,~~ ; and 7 8

6.9.1.16.h ~~[~~h. The effective plugging percentage for all plugging ~~[and tube repairs]~~ in each SG~~, and]~~ .

~~[i. Repair method utilized and the number of tubes repaired by each repair method.]~~

5.0 ADMINISTRATIVE CONTROLS

6.9 ## 5.6 Reporting Requirements

6.9.1 The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.6 5.6.1 Annual Radiological Environmental Operating Report

--------------------~~REVIEWER'S~~ NOTE--------------------
6.9.1 Note ~~[~~ A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. ~~]~~ 1

6.9.1.6
DOC L01

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01 The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[~~in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979~~]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 1

6.9.1.8 5.6.2 Radioactive Effluent Release Report

--------------------~~REVIEWER'S~~ NOTE--------------------
6.9.1 Note ~~[~~ A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. ~~]~~ 1

6.9.1.8

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

6.9.1.14 5.6.3 CORE OPERATING LIMITS REPORT

6.9.1.14 a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~[The individual specifications that address core operating limits must be referenced here.]~~ INSERT 1 1

6.9.1.14.a b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~-----------------------------REVIEWER'S NOTE-----------------------------------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~---~~ 3

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ INSERT 2 1

6.9.1.14.b c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.14.c d. The COLR, including any midcycle revisions or supplements, shall be provided ~~upon~~ within 30 days of issuance for each reload cycle to the NRC. 2

6.9.1.15 5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT

6.9.1.15 a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ INSERT 3 1

ITS 5.6

1 INSERT 1

SII

~~1. SL 2.1.1, "Reactor Core Safety Limits";~~

1 ~~2~~. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";

2 ~~3~~. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";

3 ~~4~~. LCO 3.1.5, "Shutdown Bank Insertion Limits";

4 ~~5~~. LCO 3.1.6, "Control Bank Insertion Limits";

5 ~~6~~. LCO 3.2.1, "Heat Flux Hot Channel Factor (F_Q(X, Y, Z))";

6 ~~7~~. LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$(X,Y))";

7 ~~8~~. LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";

8 ~~9~~. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," ~~,~~ $f_1(\Delta I)$ limits for Overtemperature ΔT and $f_2(\Delta I)$ limits for Overpower ΔT Nominal Trip Setpoint s ~~denoted values;~~ and

~~10. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and~~

9 ~~11~~. LCO 3.9.1, "Boron Concentration."

1 INSERT 2

1. BAW-10180P-A, Revision 1, "NEMO - Nodal Expansion Method Optimized," March 1993
2. BAW-10169P-A, Revision 0, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989
3. BAW-10163P-A, Revision 0,"Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989
4. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model," March 2001
5. BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003
6. BAW-10186-A, Revision 2, "Extended Burnup Evaluation," June 2003
7. EMF-2103P-A, Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003
8. BAW-10241P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT," July 2005
9. BAW-10199P-A, Revision 0, "The BWU Critical Heat Flux Correlations," August 1996
10. BAW-10189P-A, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," January 1996
11. BAW-10159P-A, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," August 1990 and
12. BAW-10231(P)(A), Revision 1, "COPERNIC Fuel Rod Design Computer Code" January 2004.

1 INSERT 3

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
2. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System"; and
3. LCO 3.5.2, "ECCS - Operating".

5.6 Reporting Requirements

5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

6.9.1.15.a b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~-----REVIEWER'S NOTE-----~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~----------~~ 3

~~[Identify the NRC staff approval document by date.]~~ INSERT 4 1

within 30 days of 2

6.9.1.15.b c. The PTLR shall be provided to the NRC ~~upon~~ issuance for each reactor vessel fluence period and for any revision or supplement thereto.

~~-----REVIEWER'S NOTE-----~~
~~The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~

~~2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~

~~3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~ 3

~~4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~

~~5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~

~~6. LTOP arming temperature limit development methodology.~~

1 INSERT 4

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves";

2. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation"; and

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

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5.6 Reporting Requirements

5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

~~7. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

~~8. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~ 3

3.3.10 ACTION 4

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or ~~F~~ I of LCO 3.3.~~[~~3~~]~~, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. 5 1

~~5.6.6 [Tendon Surveillance Report~~

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ 4

6.9.1.16 5.6.~~7~~ 6 Steam Generator Tube Inspection Report 4

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.~~9~~ 7, "Steam Generator (SG) Program." The report shall include: 6

6.9.1.16.a a. The scope of inspections performed on each SG~~,~~ ;

6.9.1.16.b b. Active degradation mechanisms found~~,~~ ;

6.9.1.16.c c. Nondestructive examination techniques utilized for each degradation mechanism~~,~~ ; 8

5.6 Reporting Requirements

5.6.~~7~~ 6 Steam Generator Tube Inspection Report (continued) 4

6.9.1.16.d d. Location, orientation (if linear), and measured sizes (if available) of service induced indications~~,~~ ; 8

6.9.1.16.e e. Number of tubes plugged ~~[or repaired]~~ during the inspection outage for each active degradation mechanism~~,~~ ;

6.9.1.16.f f. Total number and percentage of tubes plugged ~~[or repaired]~~ to date~~,~~ ;

6.9.1.16.g g. The results of condition monitoring, including the results of tube pulls and in-situ testing~~,~~ ; and 7 8

6.9.1.16.h ~~[~~h. The effective plugging percentage for all plugging ~~[and tube repairs]~~ in each SG~~, and]~~ .

~~[i. Repair method utilized and the number of tubes repaired by each repair method.]~~

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JUSTIFICATION FOR DEVIATIONS
ITS 5.6, STEAM GENERATOR TUBE INSPECTION REPORT

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. ISTS 5.6.6 provides requirements for the Tendon Surveillance Report. The Containment design at SQN does not include pre-stressed concrete tendons. Therefore, this report is not included in the SQN ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

5. Changes made to reflect those changes made to ITS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation."

6. Changes made to reflect those changes made to ITS 5.5.7, "Steam Generator (SG) Program."

7. Sequoyah Unit 1 and 2 are not licensed for repair of SG tubes, so the bracketed allowance has been deleted.

8. These punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications NEI 01-03, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

There are no specific No Significant Hazards Considerations for this Specification.

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ATTACHMENT 7

ITS 5.7, HIGH RADIATION AREA

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

~~6.11 RADIATION PROTECTION PROGRAM (DELETED)~~

5.7 6.12 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

5.7.1.a a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

5.7.1.b b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent~~, associated radiation survey,~~ that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

A02

5.7.1.c c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7.1.d d. Each individual or group entering such an area shall possess:

5.7.1.d.1 1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

5.7.1.d.2 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

5.7.1.d.3 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7.1.d.4 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7.1.d.4.(i) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.1.d.4.(ii) (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

5.7.1.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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A01

5.7.2 6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

5.7.2.a a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

5.7.2.a.1 1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee. s and

5.7.2.a.2 2. Doors and gates shall remain locked except ~~when needed for~~ during periods of personnel or equipment ~~access~~ entry or exit.

5.7.2.b b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent, ~~associated~~ that includes specifications of radiation ~~survey~~ dose rates in the immediate work area(s), and other appropriate radiation protection equipment and measures. A02

5.7.2.c c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7.2.d d. Each individual or group entering such an area shall possess:

5.7.2.d.1 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

5.7.2.d.2 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7.2.d.3 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7.2.d.3.(i) (i) Be under ~~the~~ surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.2.d.3.(ii) (ii) Be under ~~the~~ surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, ~~of~~ or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

5.7.2.d.4 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

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5.7.2.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2.f f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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~~6.11 RADIATION PROTECTION PROGRAM (DELETED~~

5.7 6.12 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

5.7.1.a a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

5.7.1.b b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent~~, associated radiation survey,~~ that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures. A02

5.7.1.c c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7.1.d d. Each individual or group entering such an area shall possess:

5.7.1.d.1 1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

5.7.1.d.2 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

5.7.1.d.3 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7.1.d.4 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7.1.d.4.(i) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.1.d.4.(ii) (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

5.7.1.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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A01

5.7.2 6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

5.7.2.a a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

5.7.2.a.1 1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee. s and

5.7.2.a.2 2. Doors and gates shall remain locked except ~~when needed for~~ during periods of personnel or equipment ~~access~~ entry or exit.

5.7.2.b b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent, ~~associated~~ that includes specifications of radiation ~~survey~~ dose rates in the immediate work area(s), and other appropriate radiation protection equipment and measures.

A02

5.7.2.c c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7.2.d d. Each individual or group entering such an area shall possess:

5.7.2.d.1 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

5.7.2.d.2 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7.2.d.3 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7.2.d.3.(i) (i) Be under ~~the~~ surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.2.d.3.(ii) (ii) Be under ~~the~~ surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, ~~of~~ or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

5.7.2.d.4 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

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5.7.2.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2.f f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 6.12.1.b and CTS 6.12.2.b state, in part, access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent and associated radiation survey. ITS 5.7.1.b and ITS 5.7.2.b state, in part, that access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area (s). This changes the CTS by specifying the document equivalent to the RWP shall include specification of radiation dose rates in the immediate work area(s).

The purpose of CTS 6.12.1.b and CTS 6.12.2.b is to specify the controls needed to access high radiation areas. This change is acceptable because the additional wording that the RWP equivalent includes a specification of radiation dose rates in the immediate work area(s) clarifies the requirements of an RWP. This is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

6.12 ~~[~~5.7 High Radiation Area~~]~~ 1

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

6.12.1.a a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

6.12.1.b b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.12.1.c c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

6.12.1.d d. Each individual or group entering such an area shall possess:

6.12.1.d.1 1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

6.12.1.d.2 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

6.12.1.d.3 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.12.1.d.4 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

6.12.1.d.4.(i) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7 High Radiation Area

6.12.1 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

6.12.1.d.4.(ii) (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

6.12.1.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation 2

6.12.2.a a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

6.12.2.a.1 1. All such door and gate keys shall be maintained under the administrative control of the shift ~~supervisor~~ manager, radiation protection manager, or his or her designees, and 2

6.12.2.a.2 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

6.12.2.b b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.12.2.c c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7 High Radiation Area

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 2

6.12.2.d d. Each individual or group entering such an area shall possess: 2

6.12.2.d.1 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

6.12.2.d.2 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

6.12.2.d.3 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

6.12.2.d.3.(i) (i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

6.12.2.d.3.(ii) (ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

6.12.2.d.4 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously ~~displaces~~ displays radiation dose rates in the area. 3

6.12.2.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

SII

or

5.7 High Radiation Area

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

6.12.2.f f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

5.0 ADMINISTRATIVE CONTROLS

6.12 ~~[~~5.7 High Radiation Area~~]~~ 1

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

6.12.1.a a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

6.12.1.b b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.12.1.c c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

6.12.1.d d. Each individual or group entering such an area shall possess:

6.12.1.d.1 1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

6.12.1.d.2 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

6.12.1.d.3 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.12.1.d.4 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

6.12.1.d.4.(i) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7 High Radiation Area

6.12.1 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

6.12.1.d.4.(ii) (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

6.12.1.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation 2

6.12.2.a a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

6.12.2.a.1 1. All such door and gate keys shall be maintained under the administrative control of the shift ~~supervisor~~ manager, radiation protection manager, or his or her designees, and 2

6.12.2.a.2 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

6.12.2.b b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.12.2.c c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

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5.7 High Radiation Area

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 2

6.12.2.d d. Each individual or group entering such an area shall possess: 2

6.12.2.d.1 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

6.12.2.d.2 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

6.12.2.d.3 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

6.12.2.d.3.(i) (i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

6.12.2.d.3.(ii) (ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

6.12.2.d.4 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously ~~displaces~~ displays radiation dose rates in the area. 3

6.12.2.e e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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SII

or

5.7 High Radiation Area

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source ~~of~~ from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

6.12.2.f f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Typographical/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, HIGH RADIATION AREA

There are no specific No Significant Hazards Considerations for this Specification.

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ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

Risk Informed Evaluation of Extensions to Containment Isolation Valve Completion Times (WCAP-15791)

Risk Informed Evaluation of Extensions to Containment Isolation Valve Completion Times (WCAP-15791)

1.0 Purpose

This analysis considers the Sequoyah (SQN) as-built, as-operated plant to ascertain acceptability of applying NRC endorsed topical report (TR) WCAP-15791-P-A Revision 2 "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times."

The benefit of this proposed change to the Technical Specifications is that completion time (CT) extensions will provide the operator flexibility by increasing the time to perform on-line CIV testing, maintenance or repair. Currently CIV completion time is limited to four-hours for all CIVs.

2.0 References and Acronyms

2.1 References

1. NUREG/CR-5496, "CCF Parameter Estimation 2007"
2. Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
3. Regulatory Guide 1.177, Rev. 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"
4. Regulatory Guide 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
5. WCAP-15791-P-A, Rev. 2 "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times"
6. NRC Accession Number 10027058, Generic Issue (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," August 2010
7. MDN-000-000-2010-0200, Rev. 1 - SQN PRA – "Summary Notebook"
8. MDN-000-000-2010-0202, Rev. 1 - SQN PRA – "Data Analysis"
9. MDN-000-000-2010-0203, Rev. 1 - SQN PRA – "Internal Flooding Analysis"
10. MDN-000-000-2010-0208, Rev. 2 - SQN PRA – "Quantification Notebook"
11. 1-SI-SXV-000-201.0, Rev. 017 - Surveillance Instruction – "Full Stroking of Category ‘A’ and ‘B’ Valves During Operation"
12. N2-88-400, Rev. 15 System Description – "Containment Isolation"
13. MDQ-000088-2013-000072 Rev. 0 "Risk-Informed Evaluation of Extension to Containment Isolation Valve Completion Times"
14. LTR-RAM-II-11-010, "R.G. 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Sequoyah Nuclear Plant Probabilistic Risk Assessment," March 2011

2.2 Acronyms

The following acronyms are used in this analysis:

AOT -	Allowed Outage Time
AOV -	Air Operated Valves
ASME -	American Society of Mechanical Engineers
CAFTA -	Computer Aided Fault Tree Analysis
CCF -	Common Cause Failure
CDE -	Cause Determination and Evaluation
CDF -	Core Damage Frequency
CDF_{SEIS} -	Core Damage Frequency due to Seismic Event
CDF_{SGTR}-	Core Damage Frequency due to Steam Generator Tube Rupture
CDF_{T} -	Total Core Damage Frequency (Internal & External Events)
CIV -	Containment Isolation Valve
CKV -	Check Valve
CRMP -	Configuration Risk Management Program
CT -	Completion Time
FTC -	Fail-To-Close
ICCDP -	Incremental Conditional Core Damage Probability
ICLERP -	Incremental Conditional Large Early Release Probability
IPE -	Individual Plant Examination
IPEEE -	Individual Plant Examination of External Events
ISLOCA -	Interfacing System Loss of Coolant Accident
LCO -	Limiting Condition for Operation
LERF -	Large Early Release Frequency
LLRT -	Local Leak Rate Testing
MOV -	Motor Operated Valves
MGL -	Multiple Greek Letter
NRC -	Nuclear Regulatory Commission
NRR -	Nuclear Reactor Regulation
PB_{RANDOM} -	Pipe Break - Random
PB_{SEIS} -	Pipe Break - Seismic
PRA -	Probabilistic Risk Assessment
PWROG -	Pressurized Water Reactor Owner's Group
RCS -	Reactor Coolant System
RG -	Regulatory Guide
SE -	Safety Evaluation
SG -	Steam Generator
SGTR -	Steam Generator Tube Rupture
SI -	Surveillance Instruction
SOV -	Solenoid Operated Valve
SQN -	Sequoyah Nuclear Power Plant
SRP -	Standard Review Plan
SRV -	Safety Relief Valve
SSE -	Safe Shutdown Earthquake
TR -	Topical Report
TS -	Technical Specifications
WCAP -	Westinghouse Commercial Atomic Power
WOG -	Westinghouse Owner's Group
XO -	Spurious (Transfers) Open

3.0 Assumptions and Analysis Basis

1. Before maintenance or repair is started on a containment isolation valve (CIV), it is assumed the other CIVs within the penetration are verified by Operations to be in their proper position(s).[1]
2. It is assumed that manually operated vent or drain valves located between the CIVs are verified by Operations to be in their closed position similar to assumption 1, as well as other normally closed manually operated valves connected to the penetration.
3. Manually operated vent or drains valves, if opened for LLRT (Local Leak Rate Testing), etc., are assumed to have a completion time based on the most restrictive CIV for the penetration. This is because the vent/drain lines are less than or equal to the CIV diameter, therefore, their CT is bounded by the larger valves in the associated penetration.
4. It is assumed that containment isolation valves that are locked closed have been verified closed (and locked to prevent inadvertent opening) by operations and therefore excluded from the analysis as potential to spuriously transfer open.
5. For this analysis, it is conservatively assumed that valves that are periodically opened/closed (e.g., containment purge valves) are normally opened.
6. If a seismic event greater than a SSE (Safe-Shutdown Earthquake) were to occur, it is assumed that all non-seismically qualified piping will fail, i.e., a probability of 1.0 is given. Sections of pipe between the containment isolation valve and the containment wall which is part of the break exclusion zone are excluded.[2]
7. Containment isolation valves are tested quarterly [Ref 11], additionally it is assumed there is one miscellaneous CIV actuation per year making a total of five actuations per year.[3]
8. Regardless of the completion time (CT) it is conservatively assumed the maintenance activity requires the entire length of the CT and is completed within that time.
9. For penetration configurations that have an "extra valve" i.e., not a CIV; the probability assigned for those non-CIV valves being in maintenance remains constant for all CTs. These valves are mainly recognized in the Reactor Coolant System (RCS) penetrations.[4] The assumption is that the CT on all non-CIV valves modeled in this analysis is 72 hours.
10. Only one valve within a single containment penetration can be in maintenance at a time.
11. Maintenance on a valve can be conducted in one of two ways:
    a) the valve is intact and capable of maintaining its pressure boundary function, or
    b) the valve is not intact and is not capable of maintaining its pressure boundary function.
12. When there are two or more valves of the same valve type in the same position (opened or closed) within a penetration, common cause failures (CCF) are included in the ICLERP and ΔLERF calculations.[5]

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1 This assumption eliminates the need to include the probability that the operable valves were mispositioned or transferred to the wrong position since they were last checked. This approach is consistent with WCAP-15791.

2 The piping in the break exclusion zone is more robust that the piping outside the zone. Therefore, consistent with the approach taken in WCAP 15791 it is assumed that the probability of this piping failing randomly or due to a seismic event is much lower than the piping outside the exclusion zone and is of no consequence to this analysis.

3 Same approach as taken in the WCAP.

4 The extra valve(s) provides an additional capability for the operator to isolate a penetration.

5 For cases whereby one CIV is out-of-service for repair, the second CIV of the same valve type has a dependent failure probability involving the common-cause beta factor. For cases where there are three CIVs of the same type, the dependent failure probability involves the gamma factor. The Multiple Greek Letter (MGL) common- cause methodology is used in this analysis. Note - different valve manufacturers is irrelevant to this analysis.

13. It is assumed that interfacing system LOCAs (ISLOCAs) result in core damage.
14. Similar to the Lead Plant, SQN does not have a full scope PRA; therefore, the internal and external at-power CDF is conservatively assumed to be <1.0E-4/yr. This value represents the upper bound for the total-at-power internal and external events CDF based on the acceptance guidelines in Regulatory Guide 1.174, Section 2.2.4, that indicates the plant total CDF should be less than 1.0E-04/yr if changes to a plant's licensing bases are made that can result in a small increase in plant risk.
15. For all standby systems connected to the reactor coolant system (RCS):
    - the system is considered "closed" inside containment and not actively connected to the RCS if there is a closed valve between the RCS and the inside containment CIV.
    - the system is considered "closed" outside of containment if there is an extra closed valve before the outside containment CIV, and if the piping from the RCS and the extra closed valve outside containment is qualified for high RCS pressures.
16. For systems whereby closed both inside and outside containment, the probability of a non-seismic and/or ISLOCA CDF release is extremely small and therefore excluded from the analysis due to the large number of normally closed valves available to isolate the penetration. Additionally, the likelihood of a random pipe break occurring inside and outside containment simultaneously, causing both systems inside and outside of containment to open is very small and also excluded from the analysis. Note - ALL piping between the RCS and the extra closed valve outside of containment must be qualified for high RCS pressures.
17. For RCS connections, during seismic and random pipe breaks, it is assumed that the piping fails between the CIV and the extra valve. The portion of piping between the CIV and the containment wall is part of the break exclusion zone, and therefore is assumed to remain intact while non-qualified pipe is assumed to fail. This eliminates crediting the extra valves to isolate the penetration.
18. Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of the plant charging systems, and therefore, are not considered small LOCAs or potential containment bypass pathways.
19. For all RCS connections, in which there are two valves of the same type (usually check valves), in series inside containment, before the RCS, common cause failure does not apply because the valves are operating under different conditions. The valve closer to the RCS is subject to a higher pressure than the downstream valve.
20. For the probability of an ISLOCA release portion of the ΔLERF calculations, when there is a normally open valve in the penetration, the open valve is not credited in the calculation. When assessing ISLOCA, the initiating event is the frequency of the closed valves within the path of release spuriously transferring open or rupturing, thus creating a flow path directly from the RCS to the outside atmosphere.
21. For all RCS connections that are normally operating, the probability of an ISLCOA release is not considered because the valves are already open and flow is occurring.

4.0 <u>Regulatory Acceptability and Discussion</u>

4.1 Regulatory Acceptability

The Office of Nuclear Reactor Regulation (NRR) issued a safety evaluation (SE) on WCAP-15791-P, Revision 2. The SE considers Technical Specification (TS) Limiting Conditions for Operation (LCO) that state the primary containment isolation valves (CIVs) must be operable for a given reactor mode of operation.

The SE concluded:

1. The TR (topical report) provides guidance including generic and plant-specific analyses to assist licensees in evaluating changes to CIV completions times (CTs).

2. The guidance is complementary to NRC Staff guidance provided in -

    - Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [Ref. 2]
    - Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications" [Ref.3]

Therefore, NRR stated the TR provides an acceptable basis to evaluate the proposed CIV CTs when used in conjunction with the RGs. Furthermore, with respect to the acceptance criteria associated with RG 1.177, the TR addresses Tiers 1 (Probabilistic Risk Assessment Capability and Insights) and 2 (Avoidance of Risk-Significant Plant Configurations). Tier 3 (Risk-Informed Configuration Risk Management) is not addressed by the TR and must be addressed in the plant-specific application.

4.2 Discussion

The TR provides a risk-informed justification for extending CIV CTs from 4-hours up to 168-hours for Westinghouse pressurized water reactors. For CIVs that do not demonstrate acceptable results for 168 hours, shorter CTs were evaluated in the report. A deterministic approach was used to determine the minimum containment hole size that would result in a large release from the containment atmosphere. These flow-paths are automatically given the 168-hours CT. All other penetrations were evaluated in the report using a PRA evaluation to verify what CT (i.e., less than 168-hours) is justified.

4.3 Regulatory Criteria

4.3.1 Standard Review Plan (SRP) 19.2

In accordance with SRP 19.2 a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets current regulations, unless it explicitly relates to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes increase risk (i.e., core damage frequency (CDF) or large early release frequency (LERF)), the increase(s) should be small and consistent with the intent of the Commission’s Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

4.3.2 Regulatory Guide 1.177

RG 1.177 provides an approach for plant-specific, risk-informed decision making for changes to the technical specifications. A three-tiered approach for evaluation of the risk associated with the proposed TS change follows:

Tier 1 - an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in:

- Change in the Core Damage Frequency (ΔCDF)
- Incremental Conditional Core Damage Probability (ICCDP)
- Change in Large Early Release Frequency (ΔLERF)
- Incremental Conditional Large Early Release Probability (ICLERP)

Tier 2 - identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change.

Tier 3 - provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision making process before taking equipment out-of-service prior to or during the CT.

4.4 Generic Assessment (From WCAP-15791)

WCAP-15791, "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times" documents the generic analysis performed by Westinghouse for the PWR Owner's Group (PWROG). The generic assessment of impact on risk is documented in section 8 of the WCAP.

The penetration configurations used in the analysis were developed to be as generic as possible. Some of the configurations may not exist within all plants, and/or some of the maintenance situations may or may not be viable for all plants. For plant-specific implementation of the generic analysis, the expectation of the WCAP is that all utilities determine the applicability of the CT in practice.

In the generic analysis, Table 8-1 provides the list of input parameters. The majority of the inputs used were obtained from PRA data; however, to make the analysis as generic as possible, the most limiting (e.g., highest failure rate) values were chosen from a plant-to-plant comparison. The approach used both deterministic and probabilistic inputs. A deterministic approach was used to determine the minimum containment hole size (>2 inches) that will result in a large release from the containment atmosphere. All other penetrations are evaluated on a probabilistic basis to demonstrate if a CT of 7-days is acceptable or to determine an appropriate lesser CT.

4.5 Methodology

The lead Plant followed the generic analysis, as does the SQN specific analysis. The implementation procedure followed consisted of five steps. For penetration configurations whereby completion times were less than the maximum allowable value of 168-hrs, a plant specific analysis is performed. Not all SQN penetration configurations were addressed by the WCAP of which a plant specific analysis is performed.

4.5.1 Step 1 Containment Penetration Data Collection[6]

This data was provided to the PRA Engineer and documented in calculation MDQ-000088-2013-000072. [Ref. 13]

4.5.2 Step 2 Confirmation of Analysis Input Parameters

The generic analysis documented in WCAP-15791 used a set of input parameters that were obtained from industry PRA data. To make the analysis generic, the most limiting values were chosen. A review is performed to confirm that parameters used in the SQN PRA are bounded by the generic analysis inputs. For those that are not bounded the calculation is re-performed using the SQN parameter.

4.5.3 Step 3 Grouping

Penetrations are grouped based on whether it is an open or closed system and the following attributes:

- Connected to Containment Atmosphere (Class I)
- Connected to the Reactor Coolant System (Class II)
- Connected to the Steam Generators (Class III)

4.5.4 Step 4 Identification of Small Lines

Small lines is a characterization based on the size of a hole in the primary containment that is the threshold for accident condition radionuclide large release to the environment. Note that the "Small Lines" characterization is applicable only to those penetrations connected to the containment atmosphere, i.e., Class I.

4.5.5 Step 5 Generic Match

For those penetrations that did not screen from further consideration in step 4 (i.e., 168-hr CT), a comparison of the generic penetrations/flow paths listed in sections 8.2.2 through 8.2.4 of the WCAP is made.

4.5.6 Guidelines

The ICLERP and/or ΔLERF (depending on which is more limiting) was recalculated using SQN specific parameters for CIVs with CTs less than 168-hours. The inputs were used in the appropriate ICLERP and ΔLERF equation based on the penetration Class and Group. Similar to the lead plant (Wolf Creek) analysis, two guidelines are to be followed:

- For penetrations having one normally open CIV - when more than one valve type is present, use the CT for the normally open valve. All valves in the penetration will be represented by this valve type.

[6] Steps 1, 3 and 4 are documented in Calculation MDQ-000088-2013-000072. [Ref. 13]

- For penetrations that have more than one normally open CIV - use the CT for the normally open valve with the highest probability of failing-to-close. All valves in the penetration will be represented by this valve type.

4.6 SQN Inputs / Specific and Generic

4.6.1 Discussion

The analysis involved replacing generic parameters with SQN specific parameters or updated industry data, and recalculating the probabilistic evaluation. The reason for this analysis is to determine which CIVs could be justified for longer CT relaxations in addition to those justified under the generic analysis. The approach taken in the generic analysis was conservative, and therefore, applicable to all Westinghouse Owner's Group (WOG) plants, including SQN. Where appropriate, the plant-specific analysis removes over-conservatisms.

The SQN CIVs that were unable to meet the full 168 hour CT extension under the generic analysis are identified in calculation MDQ-000088-2013-000072 [Ref. 13]. The methodology, terminology, basis and assumptions that were applicable in the generic analysis are all applicable to the SQN specific analysis. The only difference is that the SQN input parameters were used in combination with generic parameters. The analyses in WCAP sections 8.2 and 8.3 were repeated using the SQN specific-parameters to calculate ICLERP and ΔLERF. For penetration configurations that differ from the WCAP, equations were developed. The purpose of this analysis is to determine CIVs that can be justified for longer completion time (CT) relaxations (>4-hrs <168-hrs) in addition to those justified under the generic analysis.

Total CDF

Similar to the Lead Plant, SQN does not have a full scope PRA; therefore, the internal and external at-power CDF is conservatively assumed to be 1.0E-4/yr. This value represents the upper bound for the total-at-power internal and external events CDF based on the acceptance guidelines in Regulatory Guide 1.174, Section 2.2.4, that indicates the plant total CDF should be less than 1.0E-04/yr if changes to a plant's licensing bases are made that can result in a small increase in plant risk.

Seismic

SQN does not have a seismic PRA; therefore, the results documented in Table D-1, "Seismic Core-Damage Frequencies Using 2008 USGS Seismic Hazard Curves" in Generic Issue-199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" was used.[Ref 6] For SQN Units 1 & 2, the limiting frequency (5.1E-05/yr) was based on the Weakest Link Model using PGA (peak ground acceleration).

Containment Isolation Valve Treatment

For all normally-closed valves, the probability of the valve spuriously transferring open is considered. For all normally-open valves, the valves have the probability of a) failing-to-close, and b) spuriously transferring open after it has closed.

The limiting (bounding) valve in the SQN PRA for a spurious open or transfer open are the Safety Relief Valves. The SRVs provide for a steam release with a distribution rate of 2.12E-7/hr. Therefore, to determine the bounding probability that a valve will spuriously transfer open follows:

P_{topre} = 2.12E-07/hr * 4 hr CT = 8.48E-07
P_{to} = 2.12E-07/hr * 168 hrs CT = 3.56E-05

The SQN PRA does not explicitly model SOVs, therefore, SOV-FTC (9.54E-04) is used. Taken from NUREG/CR-6928.

The CIV corrective maintenance frequency (m) is derived from the highest valve failure rate used in this analysis (SOV FTC)[7] which is approximately 9.54E-04 per demand, meaning the component is expected to approximately fail every 1000 actuations [(1)/(9.54E-4) $\simeq$ 1000]. Each CIV is tested quarterly [Ref 6], and in addition it is assumed that there is one[8] miscellaneous CIV actuations per year. Therefore, giving a total of five CIV actuations per year. Dividing 1000 actuations by five actuations per year yields an approximate 200 year period per failure, or a corrective maintenance frequency of 0.005 per year. Therefore, the probability that a CIV is unavailable due to maintenance during a CT of 4 and 168 hours is calculated as follows:

Current CT 4 hrs	P_{m1} = [(4-hrs /8760-hrs/yr) * (0.005)] =	2.28E-06
Extra Valve Assumed CT 72-hrs	P_{mE} = [(72-hrs/8760-hr/yr)*(0.005)] =	4.11E-05
Extended CT 168 hrs	P_{m2} = [(168-hrs / 8760-hrs/yr) * (0.005)] =	9.59E-05

Containment Hole Size [Ref 5 Section 8.3]

Penetration flow paths connected to the containment atmosphere (this excludes RCS and SG connections) that have piping diameters smaller than a minimum value are an insufficient size to result in a large release. These penetrations automatically default to the 168-hour CT. Based on discussion with the NRC, the WOG applies a greater than 2-inch containment hole size for a large release.

4.6.2 Confirmation of Analysis Input Parameters

Table 4-1 Core Damage Frequencies

Input	Parameter	Generic Analysis	SQN-1	SQN-2
Total Core Damage Frequency/yr	CDF_T	1.00E-04	1.00E-04[Note 1]	1.00E-04[Note 1]
Core Damage Frequency Due to Seismic Event/yr	CDF_{SEIS}	4.41E-05	5.1E-05[Note 2]	5.1E-05[Note 2]

7 For SOV CIVs the generic values from the WCAP are used. SRV - Water relief are also treated independently from the generic analysis.
8 The Lead Plant analysis assumed only one additional actuation per year for a total of five. The SQN analysis applies the same assumption.

Table 4-1 Core Damage Frequencies

Input	Parameter	Generic Analysis	SQN-1	SQN-2
Core Damage Frequency/yr Due to Steam Generator Tube Rupture	CDF_{SGTR}	9.44E-06	1.75E-08[Ref 7 Tbl 6]	1.78E-08[Ref 7 Tbl 6]

Note 1: The SQN Internal Events + Internal Flooding Rev. 6 model quantifies a CDF of 1.59E-05 and 1.48E-05 [Ref 1] for Units 1 and 2, respectively. Similar to the generic analysis and lead plant analysis SQN does not have a full scope PRA; therefore, a generic value of 1.00E-04/yr is used in the analysis to represent the total CDF from internal and external events.

Note 2: The generic core damage frequency due to a seismic event (CDF_{SEIS}) per year was obtained from the results of GI-199. [Ref 6]

The seismic frequency used in the plant specific analysis is greater than that used in the generic analysis. Therefore, generic calculations that required the large release due to seismic CDF calculation are recalculated for SQN to ascertain the CT.

4.6.3 Valve Failure Probabilities, Pftc (Per Demand)

Table 4-2 - Valve Fail-To-Close and Fail-To-Reseat Probabilities

Valve Type	Failure Mode	Parameter	Generic Analysis	SQN[Note 1]
AOV[Note 2]	Fail-To-Close	$AOV_{ftc\text{-}aov}$	1.81E-02	5.76E-04
CKV	Fail-To-Close	$CKV_{ftc\text{-}ckv}$	3.44E-02	1.04E-04
MOV	Fail-To-Close	$MOV_{ftc\text{-}mov}$	1.09E-02	2.77E-04
SOV	Fail-To-Close	$SOV_{ftc\text{-}Sov}$	1.81E-02	9.54E-04
SRV - Steam[Note 3]	Fail-To-Reseat	$SRV_{ftc\text{-}srvs}$	2.50E-02	6.76E-05
SRV - Water	Fail-To-Reseat	$SRV_{ftc\text{-}srvw}$	2.50E-02	6.25E-02

As indicated in Table 4-2, the SQN inputs for valve failures (with exception of SRV-Water) are bounded by the generic analysis. Generic calculations that included water release SRVs are re-analyzed for the CT applicable to SQN.

Note 1: PRA model of Record, rev. 1, CAFTA .rr file. SOV-FTC (9.54E-04) is from NUREG/CR-6928 as SQN does not model this valve/mode in the PRA.

Note 2: AOVs are grouped into three categories, the most restrictive value is used for this analysis.

Note 3: SRVs are split into two categories, steam release (6.76E-05) and water release (6.25E-02). Table 8-1 from the WCAP [Ref 5] does not differentiate between water and steam. Judging from the value used in the generic analysis the steam and water relief SRVs may have been treated together.

4.6.4 Valve Failure Probabilities, Beta and Gamma Factors

Table 4-3 - Generic and SQN Specific Beta, Gamma Factors

Valve Type	Parameter	Generic Analysis (Valve fail-to-close, $beta_{ftc}$)	SQN [Note 1] (Valve fail-to-close, $beta_{ftc}$)
AOV	$beta_{ftc-aov}$	0.1	1.63E-02
CKV	$beta_{ftc-ckv}$	0.1	8.50E-03
MOV	$beta_{ftc-mov}$	0.088	1.54E-02
SOV	$beta_{ftc-sov}$	0.1	0.1 [Note 2]
SRV-Steam	$beta_{ftc-srvS}$	0.22	7.19E-02
SRV-Water	$beta_{ftc-srvW}$	0.22	0.22 [Note 2]
all valve types	Due to Valve Transferring Open, $beta_{to}$	0.1	0.1 [Note 2]
all valve types	Due to Valve Transferring Open, $gamma_{to}$	0.5	0.5 [Note 2]

Note 1 - The conservative generic valves for SOVs and SRV-Water failure-to-close is used in the SQN analysis.

Note 2 - The conservative generic values for the beta and gamma values are used in the SQN analysis.

The SQN parameters listed in Table 4-3 are bounded by those used in the generic analysis.

4.6.5 Spurious (Transfer) Open Probabilities and Beta Factors

Table 4-4 Spurious Open and Beta Factor Values

Parameter	Component	Description	Value	Source
AOV XO	Air-Operated Valve	Probability AOV spuriously opens per hour	1.82E-07	NUREG/CR-5496 (Section 2.4)
AOV_{BETA}		Beta factor - AOV spuriously opens	1.63E-02	
$P_{topre-aov}$		Probability AOV spuriously opens during 4-hr CT	7.28E-07	Calculated
P_{to-aov}		Probability AOV spuriously opens during 168-hr CT	3.06E-05	Calculated
CKV XO	Check Valve	Probability CKV spuriously opens per hour (Leakage)	2.96E-08	NUREG/CR-6928 (Table 5-1)

Parameter	Component	Description	Value	Source
CKV_{BETA}		Beta factor - CKV Fails to Remain Closed	3.0E-02	NUREG/CR-5496 (2.5.1.2)
$P_{topre-ckv}$		Probability CKV spuriously opens during 4-hr CT	1.18E-07	Calculated
P_{to-ckv}		Probability CKV spuriously opens during 168-hr CT	4.97E-06	Calculated
MAN XO	Manual Valve	Probability MAN spuriously opens per hour (Leak)	6.67E-08	NUREG/CR-6928 (Table 5-1)
MAN_{BETA}		Beta factor - MAN spuriously opens	0.1	WCAP (Generic)
$P_{topre-man}$		Probability MAN spuriously opens during 4-hr CT	2.67E-07	Calculated
P_{to-man}		Probability MAN spuriously opens during 168-hr CT	1.12E-05	Calculated
MOV XO	Motor-Operated Valve	Probability MOV spuriously opens per hour	4.45E-08	NUREG/CR-6928 (Table 5-1)
MOV_{BETA}		Beta factor - MOV spuriously opens	2.67E-02	NUREG/CR-5496 (2.3.1.2)
$P_{topre-mov}$		Probability MOV spuriously opens during 4-hr CT	1.78E-07	Calculated
P_{to-mov}		Probability MOV spuriously opens during 168-hr CT	7.48E-06	Calculated
SRV XO	Safety Relief Valve	Probability SRV (Water or Steam Release) spuriously opens per hour	2.12E-07	Calculated
SRV_{BETA-W}		Beta factor - SRV-Water fails to reseat after opening	0.22	WCAP
SRV_{BETA-S}		Beta factor - SRV-Water fails to reseat after opening	0.22	WCAP
$P_{topre-srv}$		Probability SRV-Water or Steam spuriously opens during 4-hr CT	8.48E-07	Section 4.6.1
P_{to-srv}		Probability SRV-Water or Steam spuriously opens during 168-hr CT	3.56E-05	Section 4.6.1
SOV XO	Solenoid Operated Valve	Probability SOV spuriously opens per hour	9.23E-08	NUREG/CR-6928 (Table 5-1)

Parameter	Component	Description	Value	Source
SOV_{BETA}		Beta factor - SOV spuriously opens	0.1	WCAP (Generic)
$P_{topre-sov}$		Probability SOV spuriously opens during 4-hr CT	3.69E-07	Calculated
P_{to-sov}		Probability SOV spuriously opens during 168-hr CT	1.55E-06	Calculated

The values listed in Table 4-4 are bounded by those used in the generic analysis.

4.6.6 Additional Inputs

Table 4-5 Additional Inputs

Parameter	Description	Generic Value	Value	Source
PB_{SEIS}	Seismic Pipe Break Probability for Non-Seismically Qualified Pipe	1.0	1.0	Assumed
PB_{RANDOM}	Random Pipe Break Frequency (per year)[1]	1.10E-03	3.14E-03	Ref. 9 Table E-1
P_{topre}	Probability that Valve Spuriously Transfers Open During 4-Hr CT (most limiting valve)	4.00E-06	8.48E-07	Calculated Section 4.6.1
P_{to}	Probability that Valve Spuriously Transfers Open During 168-Hr CT (most limiting valve)	1.68E-04	3.56E-05	Calculated Section 4.6.1
P_{m1}	Probability that a CIV is Disabled due to Maintenance (per demand) during a 4-hr CT	4.00E-06	2.28E-06	Calculated Section 4.6.1
P_{mE}	Probability that Extra Valve is Disabled due to Maintenance (per demand) [assume extra valve currently has 72 hour CT]	Any Valve	4.11E-05	Calculated Section 4.6.1
P_{m2}	Probability that a CIV is Disabled due to Maintenance (per demand) during a 168-hr CT	Any Valve	9.59E-05	Calculated Section 4.6.1

Note 1 - The random (passive) pipe break frequency is based on the most limiting frequency used in the SQN internal flooding analysis.

The values listed in Table 4-5 are bounded by those used in the generic analysis with exception to the random pipe break frequency. Therefore, the generic analysis that include the random pipe break frequency are reanalyzed for SQN to determine the applicable CT.

4.7 Application Tiers 1, 2 and 3

4.7.1 Tier 1 PRA Applicability and Insights

The SQN PRA was subjected to a full scope Peer Review in accordance with R.G. 1.200 [Ref 4] requirements in 2011. The conclusions of the peer review team follow: [Ref 13]

- The overall model structure is robust and well-developed, but needs refinement,
- Documentation is thorough, detailed and well-organized such that comparison with the standard is facilitated,
- The process and tools utilized are at the state-of-technology and generally consistent with Capability Category II, and
- The PRA maintenance and update program includes all necessary processes and does a very good job of tracking pending changes.

The following areas are addressed:

1. Assurance that the plant-specific PRA reflects the as-built, as-operated plant.

   A key attribute to the ASME/ANS Standard is to assess how the PRA modeled the as-built as-operated plant. The Data Analysis technical element addresses this. The PRA model at the time of the Peer Review was judged to meet this requirement, and HLR-MU-B stated that a PRA configuration control process is in place, and governed by procedure which provides a reasonable assurance that the as-built, as-operated planted is reflected through routine maintenance and upgrades to the PRA.

2. Assurance that the applicable PRA updates include the findings from the individual plant evaluation (IPE) and the IPE for External Events.

   The SQN PRA has been updated multiple times since the completion of the IPE and IPEEE from the 1990’s. The technical adequacy of the PRA was established by Peer Review in early 2011. The current model of record represents a significantly more mature PRA as compared to the IPE and IPEEE.

3. Assurance that conclusions from the peer review, including facts and observations that are applicable to this application have been resolved.

   For areas, whereby the Peer Review determined work was necessary to meet Capability Category II a Facts & Observation (F&O) was initiated. The resolutions to the F&Os are documented in the PRA Summary Notebook. [Ref 7]

4. Assurance that there is PRA configuration control and updating, including PRA quality assurance programs, associated procedures, and PRA revision schedules.

   TVA procedure NPG-SPP-09.11, Probabilistic Risk Assessment (PRA) Program, covers the management of PRA applications and periodic PRA updates. Periodic changes made to the base plant-specific PRA model are required to incorporate system, structure, component and operating philosophy changes, and new plant-specific data.

5. Assurance that there is PRA adequacy, completeness, and applicability with respect to evaluating the risk associated with the proposed CIV CT extensions.

   SQN specific parameters and PRA results applicable to the proposed risk-informed application of CIV completion time extensions are well documented in references 7 through 10. The PRA has been subjected to a Peer Review in early 2011 that assessed the technical adequacy of the SQN PRA.

6. Assurance that plant design or operational modifications that are related to or could impact the proposed CT extensions are reflected in the PRA revision used in the plant-specific application, or a justification for not including those modifications in the PRA.

   In accordance with TVA procedure NPG-SPP-09.11, "Probabilistic Risk Assessment (RPA) Program," plant modifications or design changes that result in new configurations, alignments, and capabilities of plant system are assessed for inclusion in model updates. Furthermore TVA procedure NEDP-26 "Probabilistic Risk Assessment (PRA)" provides the requirements for the cumulative impact of plant configuration changes, including plant-specific design, procedure and operational changes that require an update to the Model of Record.

4.7.2 Tier 2 Avoidance of Risk-Significant Plant Configurations

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The process SQN uses to avoid risk-significant plant configurations is governed by TVA procedure NPG-SPP-07.1, "On-Line ~~Risk~~ Work Management." The procedure applies to all work activities that affect or have the potential to affect a plant component, system, or unit configuration. A risk assessment methodology is used for on-line maintenance and shutdown operations. For on-line maintenance, a risk assessment is performed prior to implementation and emergent work is evaluated against the assessed scope. Shutdown risk is assessed in accordance with TVA procedure NPG-SPP-07.2, "Outage Management." Furthermore, TVA procedure NPG-SPP-07.3, "Work Activity Risk Management Process" provides an integrated process for assessing and reducing the likelihood and/or consequences of an adverse event. SQN employs a work management process that utilizes Functional Equipment Groups (FEGs). The grouping qualitatively assessed work activities and components and made logic ties that prevent certain risk-significant plant configurations for being scheduled simultaneously.

4.7.3 Tier 3 Risk-Informed Configuration Risk Management

In accordance with the requirements of 10CFR50.65(a)(4) SQN assesses and manages plant configurations prior to taking the maintenance configuration. The proposed plant configuration is modeled in the computer code EOOS (Equipment Out Of Service) to determine the change in the core damage frequency (CDF) and the large early release frequency (LERF). The initial risk assessment is performed six - nine weeks prior to implementation to allow for risk-informed

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sequencing of activities as necessary and for other actions determined based on risk insights gleaned from the initial assessment. The well defined process is governed by TVA procedure NPG-SPP-07.1, "On-Line ~~Risk~~ Work Management." The quantified change in risk is used as one input with respect to configuration risk management. Furthermore, the process prescribes successive higher levels of management approval for plant configurations resulting in an increase in risk at various levels. Although not quantified, work management compensatory measures are prescribed as the risk level increases to limit the likelihood of entering an unplanned configuration (i.e., protected trains/equipment) or to limit the consequences of an

unattended action. Outage Risk Management is controlled in accordance with TVA procedure NPG-SPP-7.2.11.

4.8 SQN Specific Analysis

4.8.1 Fault Trees and Applicable Penetrations

The following fault trees were developing to calculate the given penetration configuration.

Table 4-6 Class I Classification and Penetrations:

SQN FT ID	Class / Group Calculation Number	Applicable Penetrations
I_A-1_4 I_A-1_168	I, A#1 I,B#3 I,C#3	X-79A X-79B X-80 X-82 X-83
I_A-3_4 I_A-3_168	I,A#3 I,B#5 I, C#5	X-4 X-5 X-6 X-7 X-9A X-9B X-10A X-10B X-11 X-29 X-43A X-43B X-43C X-43D X-47A X-47B X-50A X-52 X-57 X-58 X-59 X-60 X-61 X-62 X-63
I_A-4-4 I_A-4-168	I,A#4	X-42 X-50B X-51 X-78 X-111 X-112 X-113
I_B-1-4 I_B-1-168	I,B#1	X-35 X-88 X-117 X-118
I_B-4-4 I_B-4-168	I,B#4	X-40D(U1) X-40D(U2) X-48A X-48B X-49A X-49B
I_B-5-4 I_B-5-168	I,B#5	X-46
I_B-6-4 I_B-6-168	I,B#6	X-53
I_C-1-4 I_C-1-168	I,C#1	X-19A X-19B

Table 4-7 Class II Classification and Penetrations:

SQN FT ID	Class / Group Calculation Number	Applicable Penetrations
II_A-6-4 II_A-6-168	II,A#6	X-20A X-20B
II_A-17-4 II_A-17-168	II,A#17	X-17 X-21 X-32
II_B-2-4 II_B-2-168	II,B#2	X-15
II_B-X44-4 II_B-X44-168	II,B#X44	X-44
N/A	II-Type A - Bounding	X-22 X-33 X-107
N/A	II-Type B - Bounding	None

Table 4-8 Class III Classification and Penetrations:

SQN FT ID	Class / Group Calculation Number	Applicable Penetrations
III_A-X12-4 III_A-X12-8	III,A#X12	X-12B X-12C
III_A-X13-4 III_A-X13-8	III,A#X13	X-13A X-13B X-13C X-13D
III_A-X14-4 III_A-X14-8	III,A#14	X-14A X-14B X-14C X-14D
N/A	III-Type A - Bounding	X-12A X-12D
N/A	III-Type B - Bounding	None

Table 4-9 All Classes - Penetrations With CT Extensions Based on WCAP Generic Analyses

Applicable Penetrations
X-16 X-24 X-25C X-26C X-27D X-86A X-86B X-86C X-102 X-104

Table 4-10 Penetrations with One or More Valves Crediting the Small Line Exclusion

Applicable Penetrations
X-4 X-5 X-6 X-7 X-9A X-9B X-10A X-10B X-11 X-15 X-16 X-17 X-19A X-19B X-20A
X-20B X-21 X-22 X-23 X-24 X-25A(U1) X-25A(U2) X-25B(U1) X-25B(U2) X-25D(U1) X-25D(U2)
X-26A X-26B(U1) X-26B(U2) X-27A X-27B X-27C X-29 X-30 X-32 X-33 X-34(U1) X-34(U2)
X-35 X-39A X-39B X-41 X-42 X-43A X-43B X-43C X-43D X-45 X-46 X-47A X-47B X-48A
X-48B X-49A X-49B X-50A X-50B X-51 X-52 X-56 X-57 X-58 X-59 X-60 X-61 X-62 X-63
X-64 X-65 X-66 X-67 X-68(U2) X-69(U2) X-70(U2) X-71(U2) X-72(U2) X-73(U2) X-74(U2)
X-75(U2) X-76(U1) X-76(U2) X-77 X-78 X-80 X-82 X-83 X-84A X-85B X-85C X-87B X-87D
X-90(U1) X-90(U2) X-91 X-92A(U1) X-92A(U2) X-92B(U1) X-92B(U2) X-93(U1) X-93(U2)
X-94A X-94B X-94C X-95A X-95B X-95C X-96C(U1) X-96C(U2) X-97 X-98 X-99(U1) X-99(U2)
X-100(U1) X-100(U2) X-101 X-102 X-103 X-104 X-106 X-107 X-111 X-112 X-113 X-114
X-115 X-116A

Table 4-11 Penetrations with No Generic Fit - Not Analyzed By PRA

Applicable Penetrations
X-40A X-40B X-108 X-109

4.8.2 Calculations / Inputs

Two calculations are performed to determine the acceptability of the extended CT. The Incremental Conditional Large Early Release Probability (ICLERP) is based on R.G. 1.177 [Ref 3] acceptable criteria of <5.0E-08. The change in the large early release frequency (ΔLERF) is based on the R.G. 1.174 [Ref 2] acceptance criteria of <1.0E-07/yr.

a) **Class I** - penetrations connected to the containment atmosphere; a failure to isolate a penetration would result in a release path to the environment. Four types:

- Type A- Flow paths connected directly to containment atmosphere and the outside environment.
- Type B- Flow paths closed inside containment and connected directly to the outside environment.
- Type C - Flow paths connected directly to containment atmosphere and closed outside containment.
- Type D - Flow paths closed inside containment and closed outside containment.

Class I Penetrations - Flow Paths Connected to the Containment Atmosphere

Release Type	Details	Comment	Input(s)
Non-seismic CDF Release	For open systems a direct connection from inside to outside containment is possible given a failure to isolate the penetration.	Release due to an internal event CDF. If core damage occurs simultaneous with a failure to isolate (spurious open, fail-to-close, etc.) the penetration a large release could occur.	• CDF_T • CT • Valve Failure Probability
Seismic CDF Release	For closed systems (either inside or outside of containment), a seismic-induced core damage event, the assumption is made that the closed loop system piping fails.	Closed systems - seismic event breaches both sides of containment. If this were to occur simultaneous with failure to isolate the penetration, an open pathway to the environment would exist.	• CDF_{seis} • CT • PB_{SEIS} • Valve Failure Probability
Random Pipe Break CDF Release	The configuration would be based on the system being open on one side of containment and the other closed.	Release due to an internal event and a random pipe break. A random pipe break of the closed system simultaneous with a failure to isolate the penetration would present a flow path to the environment.	• CDF_T • CT • PB_{RANDOM} • Valve Failure Probability

b) **Class II -** penetration flow paths connected to the reactor coolant system. Two types:

- Type A - Standby system flow paths.
- Type B - Normally operating system flow paths.

Class II Penetrations - Flow Paths Connected to the Reactor Coolant System

Release Type	Details	Comment	Input(s)
Non-seismic CDF Release	Connected to the RCS and open outside containment.	Release due to an internal event CDF. Core damage simultaneous with CIV failure to isolate (spurious open, fail-to-close, etc.) the penetration a large release could occur.	• CDF_T • CT • Valve Failure Probability
Seismic CDF Release	For system connected to the RCS and open outside it is assumed that a seismic event results in core damage.	Release due to seismic event resulting in core damage whereby all closed loop piping fails both inside and outside containment simultaneous with CIV failure creating an opening to the environment.	• CDF_{SEIS} • CT • PB_{SEIS} • Valve Failure Probability

Release Type	Details	Comment	Input(s)
Random Pipe Break CDF Release	For systems connected to the RCS and open outside containment.	Release due to an internal event and a random pipe break. If core damage were to occur simultaneous with a random pipe break inside containment and a failure to isolate the penetration, the system would no longer be connected to the RCS, therefore, allowing an open flow path to the environment.	• CDF_T • CT • PB_{RANDOM} • Valve Failure Probability
Interfacing System Loss of Coolant Accident (ISLOCA)	For standby systems connected to the RCS and open outside containment.	Release due to containment bypass, if the CIVs fail (CIV failure is the initiator), an ISLOCA would occur resulting in core damage.	• Valve Failure Probability

c) **Class III** - penetrations with flow paths connected to the steam generators. Two types:

- Type A- Flow paths connected to the steam generator secondary side and open to the outside environment.
- Type B- Flow paths connected to the steam generator secondary side and closed to the outside environment.

Class III Penetrations - Flow Paths Connected to the Steam Generator Secondary Side

Release Type	Details	Comment	Input(s)
Seismic CDF Release	For systems connected to the SG secondary side and open or closed outside containment.	Release due to seismic event resulting in core damage whereby all closed loop piping fails both inside and outside containment simultaneous with CIV failure creating an opening to the environment.	• CDF_{SEIS} • CT • PB_{SEIS} • Valve Failure Probability
Random Pipe Break CDF Release	For systems connected to the SG secondary side and open outside containment.	Release due to an internal event and a random pipe break. If core damage were to occur simultaneous with a random pipe break inside containment and a failure to isolate the penetration, the system would no longer be connected to the SGs, therefore, allowing an open flow path to the environment.	• CDF_T • CT • PB_{RANDOM} • Valve Failure Probability

Release Type	Details	Comment	Input(s)
Steam Generator Tube Rupture (SGTR)	For systems connected to the steam generator secondary side and open to the outside atmosphere.	Release due to SGTR simultaneous with a core damage event and failure to isolate the penetration which would result in an open pathway to the environment	• CDF_{SGTR} • Valve Failure Probability • CT
Steam Generator Tube Rupture (SGTR) With Random Pipe Break	For systems connected to the SG secondary side and a closed systems outside containment.	Random pipe break outside of containment followed by a SGTR and CIV failure would result in an open path to the environment.	• CDF_{SGTR} • PB_{RANDOM} • Valve Failure Probability • CT

5.0 Results

The results of the generic analysis and the SQN specific analysis are recorded in Attachment 1. Many CIVs were justified at 168-hr CTs based on application of the generic analysis. Justification could not be made for some CIVs, therefore their CTs remain at 4-hrs.

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-4	30-56	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-57	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-555TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-5	30-58	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-59	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-554TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-6	30-50	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	30-51	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-6 (cont)	30-558TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-7	30-52	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-53	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-557TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-9A	30-8	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-7	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-563TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-9B	30-10	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-9B (cont)	30-9	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-562TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-10A	30-15	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-14	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-561TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-10B	30-17	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	30-16	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-560TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-11	30-20	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-19	This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Assume more limiting condition of valve open. Same valve type IC and OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-559TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release. Normally closed valve.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-12A	3-33	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.3, MFIV.	N/A				
	3-164	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
	3-164A	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
	3-174	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	3-904	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-903	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-857	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-12A (cont)	3-889	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-849	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-853	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	U2 ONLY 2-3-504	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-12B	3-47	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.3, MFIV.	N/A				
	3-502	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC. Drain valve assume same CT as 3-47.	III-A-12BC	System pressure boundary maintained System pressure boundary compromised	8-hrs 8-hrs	8-hrs 8-hrs	8-hrs 8-hrs
X-12C	3-87	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.3, MFIV.	N/A				

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	3-500	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC. Drain valve assume same CT as 3-87.	III-A-12BC	System pressure boundary maintained System pressure boundary compromised	8-hrs 8-hrs	8-hrs 8-hrs	8-hrs 8-hrs
X-12D	3-100	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.3, MFIV.	N/A				
	3-171	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
	3-171A	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
X-12D (cont)	3-175	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
	3-907	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-906	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-858	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-890	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-850	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	3-854	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	UNIT 2 ONLY 2-3-506	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III,A Bounding	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-13A	1-5	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.4, ARV.	N/A				
	1-4	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.2, MSIV.	N/A				
	1-147	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-13A (cont)	1-15	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.5, AFW.	N/A				
	1-522	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-523	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-524	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-525	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-526	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-922	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	1-536	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-13B	1-12	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.4, ARV.	N/A				
	1-11	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.2, MSIV.	N/A				
	1-148	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-13B (cont)	1-517	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-518	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-519	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-520	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-521	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-923	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-534	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-13C	1-23	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.4, ARV.	N/A				
	1-22	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.2, MSIV.	N/A				
	1-149	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-512	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
X-13C (cont)	1-513	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-514	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-515	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-516	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-924	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-532	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-13D	1-30	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.4, ARV.	N/A				

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	1-16	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.3, MFIV.	N/A				
	1-150	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-29	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.2, MSIV.	N/A				
	1-527	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
X-13D (cont)	1-528	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-529	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-530	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-531	THIS VALVE IS NOT COVERED BY ITS SECTION 3.6.3. IT IS COVERED BY ITS SECTION 3.7.1, MSSV.	N/A				
	1-925	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-538	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-13	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-14A	1-182	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve IC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	1-14	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	43-58	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-825	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-14B	1-184	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve IC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-14B (cont)	1-32	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	43-64	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-827	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-14C	1-183	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve IC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-25	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	43-61	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-826	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-14D	1-181	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve IC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-7	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	43-55	Direct connection to Steam Generator. Closed system IC to open system OC. Normally open valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
	1-824	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed valve OC.	III-A-14	System pressure boundary maintained System pressure boundary compromised	8-Hrs 8-Hrs	8-Hrs 8-Hrs	8-Hrs 8-Hrs
X-15	62-72	Normally operating system; RCS connection. 3 Valves IC, 1 normally open, 2 normally closed. 1 Valves OC, normally open. All Valves the same type. This valve normally closed IC.	II-B-2	System pressure boundary maintained System pressure boundary compromised	168-Hrs 168-Hrs	168-Hrs 168-Hrs	168-Hrs 168-Hrs
	62-73	Normally operating system; RCS connection. 3 Valves IC, 1 normally open, 2 normally closed. 1 Valves OC, normally open. All Valves the same type. This valve normally open IC.	II-B-2	System pressure boundary maintained System pressure boundary compromised	168-Hrs 168-Hrs	168-Hrs 168-Hrs	168-Hrs 168-Hrs
	62-74	Normally operating system; RCS connection. 3 Valves IC, 1 normally open, 2 normally closed. 1 Valves OC, normally open. All Valves the same type. This valve normally closed IC.	II-B-2	System pressure boundary maintained System pressure boundary compromised	168-Hrs 168-Hrs	168-Hrs 168-Hrs	168-Hrs 168-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-662	Normally operating system; This valve is a PRESSURE RELIEF VALVE, WHICH RELIEVES TO THE PRESSURIZER RELIEF TANK INSIDE CONTAINMENT and is not directly connected to the RCS, Given this scenario, flow path is also smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-77	Normally operating system; RCS connection. 3 Valves IC, 1 normally open, 2 normally closed. 1 Valves OC, normally open. All Valves the same type. This valve normally open OC.	II-B-2	System pressure boundary maintained System pressure boundary compromised	168-Hrs 168-Hrs	168-Hrs 168-Hrs	168-Hrs 168-Hrs
	62-707	This is a normally closed test valve which communicates with containment atmosphere not RCS for flow thru valve from IC to OC. Therefore flow path is smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-16	62-543	Normally operating system; RCS connection. 1 CIV IC normally open - 1 CIV OC, normally open - different valve types (The normally open check valve IC has another normally open check valve in series between it and the RCS) (The normally open CIV OC has another normally open valve downstream of it, same valve type)	II,B #3	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-90	Normally operating system; RCS connection. 1 CIV IC normally open - 1 CIV OC, normally open - different valve types (The normally open check valve IC has another normally open check valve in series between it and the RCS) (The normally open CIV OC has another normally open valve downstream of it, same valve type)	II,B #3	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-709	Normally operating system; RCS connection. Continues to operate during accident, therefore not considered a path for release directly from RCS since flow continues to be forced into RCS; therefore, release scenario is from containment atmosphere, flow path is smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-544	This is a normally closed test valve which communicates with containment atmosphere not RCS for flow thru valve from IC to OC. Therefore flow path is smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-17	63-640	RCS connection; standby system. Open system IC. Closed system OC. Normally closed valve IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-643	RCS connection; standby system. Open system IC. Closed system OC. Normally closed valve IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
X-17 (cont)	63-158	No direct connection to RCS piping; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-172	RCS connection; standby system. Open system IC. Closed system OC. Normally closed valve IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-637	No direct connection to RCS piping; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-636	No direct connection to RCS; Only release path is from containment atmosphere to environment via RHA. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-642	No direct connection to RCS; Only release path is from containment atmosphere to environment via RHA. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-870	No direct connection to RCS; Only release path is from containment atmosphere to environment via RHA. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-19A	63-72	Standby system. Containment atmosphere at sump. Closed system OC. 1 valve - normally closed (OC or IC)	I-C-1	System pressure boundary maintained Pressure Boundary Compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-593	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-591	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-19B	63-73	Standby system. Containment atmosphere at sump. Closed system OC. 1 valve - normally closed (OC or IC)	I-C-1	System pressure boundary maintained Pressure Boundary Compromised	168-hrs 4-hrs	168-hrs 168-hrs	168-hrs 4-hrs
	63-592	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-590	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-20A	63-112	No direction connection to RCS piping; Line isolated by 2 normally closed valves. Valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-635	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	63-633	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	63-94	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	63-631	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-20A (cont)	63-667	No direction connection to RCS piping; valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-661	No direction connection to RCS piping; valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-833	No direction connection to RCS piping; valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-20B	63-111	No direction connection to RCS piping; Line isolated by 2 normally closed valves. Valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-632	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	63-634	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-20B (cont)	63-93	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	63-630	RCS connection; standby system. 2 check valves IC each have another normally closed check valve in series. 2 valves OC in parallel. 1 normally open and 1 normally closed.	II-A-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-413	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-659	No direction connection to RCS piping; valve is IC; only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-660	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-21	63-167	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-547	RCS connection; standby system; normally closed check valve IC; another check valve upstream IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-549	RCS connection; standby system; normally closed check valve IC; another check valve upstream IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
X-21 (cont)	63-157	No Direct connection to RCS; Open system IC to closed system OC; Normally closed valve OC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-648	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-649	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-650	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-862	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-313A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-314A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-21 (cont)	63-317A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-318A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-22	63-174	Connected to RCS accumulators thru another normally closed FCV. Not directly connected to RCS.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-581	RCS connection; standby system. Open system IC and closed system OC. Normally closed valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-25 (FCV)	RCS connection; standby system. Open system IC and closed system OC. Normally closed valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-26 (FCV)	RCS connection; standby system. Open system IC and closed system OC. Normally closed valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	Unit 2 Only 2-63-816	Not directly connected to RCS. Drain or vent valve. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-25 (FSV)	Not directly connected to RCS. Open system IC to closed system OC. Normally closed valve OC.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-26 (FSV)	Not directly connected to RCS. Open system IC to closed system OC. Normally closed valve OC.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-23	43-310	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-309	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	43-492	Test valve connected to containment atmosphere IC. Only release path is from containment atmosphere to environment via the sampling system. Source piping is 3/8" and smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-24	68-559	No direct connection to RCS; penetration flow path connects open system IC to closed system OC. Normally closed valve IC & OC. Different valve types. This valve IC.	I,A#6	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-505	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-512	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-513	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-511	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-24 (cont)	63-536	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-535	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-534	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-626	No direct connection to RCS; penetration flow path connects open system IC to closed system OC. Normally closed valve IC & OC. Different valve types. This valve OC.	I,A#6	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-627	No direct connection to RCS; penetration flow path connects open system IC to closed system OC. Normally closed valve IC & OC. Different valve types. This valve OC.	I,A#6	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	68-560	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	68-561	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-517	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-518	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-638	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-25A	43-2	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	43-3	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-25B	-	THE DP SENSORS ARE CLOSED SYSTEMS OUTSIDE OF CONTAINMENT THAT ARE ATTACHED DIRECTLY TO CONTAINMENT. NO ISOLATION VALVES ARE EMPLOYED FOR THESE SENSORS AS THEY USE A DOUBLE DIAPHRAGM SYSTEM FOR DP MEASUREMENT. THE DIAPHRAGMS ARE QUALIFIED FOR POST-LOCA USE. NO DIRECT CONNECTION TO RCS. FLOW PATH SMALLEER THAN THAT REQUIRED TO RESULT IN A LARGE RELEASE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-311Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-311X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-44Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-25B (cont)	30-44X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-25C	-	THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-25D	43-11	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-12	Lines connected to the RCS 3/8" in diameter or less are within the makeup capability of plant's charging systems and therefore, are not considered small LOCAs or potential containment bypass pathways. (Sec 8.2.3 of WCAP)	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-26A	-	THE DP SENSORS ARE CLOSED SYSTEMS OUTSIDE OF CONTAINMENT THAT ARE ATTACHED DIRECTLY TO CONTAINMENT. NO ISOLATION VALVES ARE EMPLOYED FOR THESE SENSORS AS THEY USE A DOUBLE DIAPHRAGM SYSTEM FOR DP MEASUREMENT. THE DIAPHRAGMS ARE QUALIFIED FOR POST-LOCA USE. NO DIRECT CONNECTION TO RCS. FLOW PATH SMALLEER THAN THAT REQUIRED TO RESULT IN A LARGE RELEASE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-310Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-310X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-43Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-43X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-26B Unit 1 Only	32-102	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-297	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-295	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-26B Unit 1 Only (cont)	32-292	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-26B Unit 2 Only	32-103	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-348	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-341	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-345	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-26C	-	THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-27A	-	THE DP SENSORS ARE CLOSED SYSTEMS OUTSIDE OF CONTAINMENT THAT ARE ATTACHED DIRECTLY TO CONTAINMENT. NO ISOLATION VALVES ARE EMPLOYED FOR THESE SENSORS AS THEY USE A DOUBLE DIAPHRAGM SYSTEM FOR DP MEASUREMENT. THE DIAPHRAGMS ARE QUALIFIED FOR POST-LOCA USE. NO DIRECT CONNECTION TO RCS. FLOW PATH SMALLEER THAN THAT REQUIRED TO RESULT IN A LARGE RELEASE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-30CX	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-30CY	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-27B	-	THE DP SENSORS ARE CLOSED SYSTEMS OUTSIDE OF CONTAINMENT THAT ARE ATTACHED DIRECTLY TO CONTAINMENT. NO ISOLATION VALVES ARE EMPLOYED FOR THESE SENSORS AS THEY USE A DOUBLE DIAPHRAGM SYSTEM FOR DP MEASUREMENT. THE DIAPHRAGMS ARE QUALIFIED FOR POST-LOCA USE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-42Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-42X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-27C	52-504	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-27C (cont)	52-505	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	52-510	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-27D	-	THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-29	70-89	No direct connection to RCS; penetration flow path connects open system IC to open system OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-698	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-92	No direct connection to RCS; penetration flow path connects open system IC to open system OC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	70-735	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-30	63-71	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-84	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-23	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-537	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-344A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-32	63-21	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-545	RCS connection; standby system; normally closed check valve IC; another check valve upstream IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-543	RCS connection; standby system; normally closed check valve IC; another check valve upstream IC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs
	63-156	No Direct connection to RCS; Open system IC to closed system OC; Normally closed valve OC.	II-A-X17	System pressure boundary maintained System pressure boundary compromised	168-hrs 4-hrs	168-hrs 4-hrs	168-hrs 4-hrs

SQN Containment Isolation Valve Completion Time Results

SQN Pent #	SQN Valve	Grouping Explanation	Group & Calc # (Note 1)	Maintenance Activity Type	ICLERP @ CT:	ΔLERF @ CT:	Justified CT
	63-541	No direct connection to RCS; isolated from RCS by double check valves. Open system IC to closed system OC. Normally closed valve OC. Drain or vent valve. Assume same CT as the shortest CT of other valves in the penetration.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-32 (cont)	63-823	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-657	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-658	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-864	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-315A	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-316A	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-311A	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-32 (cont)	63-612A	No direct connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-33	63-551	RCS connection; standby system; open system IC to closed system OC. Normally closed valve IC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-553	RCS connection; standby system; open system IC to closed system OC. Normally closed valve IC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-555	RCS connection; standby system; open system IC to closed system OC. Normally closed valve IC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-557	RCS connection; standby system; open system IC to closed system OC. Normally closed valve IC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-121	No direction connection to RCS, standby system. Isolated from RCS by at least 2 normally closed valves. Open system IC to closed system OC. Normally closed valve IC. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-22	Penetration flow path connects OPEN system IC to closed system OC. Normally open valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	63-653	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-33 (cont)	63-654	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-655	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-836	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-656	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-831	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-325A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-326A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-33 (cont)	63-319A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-320A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-321A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-322A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-323A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	63-324A	No direction connection to RCS piping; Valve is IC; Only release path is from containment atmosphere to environment via the SIS system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-34 Unit 1 Only	32-377	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-110	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-34 Unit 1 Only (cont)	32-375	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-373	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-34 Unit 2 Only	32-387	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-111	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-385	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-383	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-35	70-85	No direct connection to RCS; penetration flow path connects closed system IC to open system OC.	I-B-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-703	No direct connection to RCS; penetration flow path connects closed system IC to open system OC.	I-B-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	70-702C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-762	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-702F	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-764	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-35 (cont)	Unit 2 Only 2-70-759	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-39A	77-868	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	63-64	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-867	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-39B	77-849	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	68-305	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-848	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-40A	3-156	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-156A	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-173	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-860	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
X-40A (cont)	3-899	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-900	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-852	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-848	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-888	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-901	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-40B	3-148	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-148A	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-172	THIS VALVE IS NOT COVERED BY TECH SPEC 3.6.3. IT IS COVERED BY TECH SPEC SECTION 3.7.5, AFW.	N/A				
	3-859	Flow thru this valves is from containment atmosphere IC to OC. Flow path smaller in size than that required to result in a large release.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
X-40B (cont)	3-842	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-897	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-896	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-855	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-847	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
	3-851	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	3-887	Direct connection to Steam Generator. Closed system IC to open system OC. Normally closed drain/vent valve OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4 hrs 4 hrs	4 hrs 4 hrs	4 hrs 4 hrs
X-40D	BLF	No direct connection to RCS; penetration flow path connects open system IC to open system OC.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-41	77-127	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-128	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-42	81-502	No direct connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve IC & OC. Different valve type. This valve IC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-42 (cont)	81-12	No direct connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve OC & IC. Different valve type. This valve OC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	81-529	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43A	62-563	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-550	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-549	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-546	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-578	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-555	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43A (cont)	62-571	No direct connection to RCS; Connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-575	No direct connection to RCS; Connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43B	62-561	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-550	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-549	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43B (cont)	62-546	No direct connection to RCS; Normally operating system, continues to operate during accident. Therefore, not considered a path for release directly for RCS since flow continues to be forced into RCS; therefore release scenario is from containment atmosphere.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-577	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-555	No direct connection to RCS; Normally closed vent/drain valve. Isolated from RCS by double check valves. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-569	No direct connection to RCS; Vent/Drain connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-573	No direct connection to RCS; Vent/Drain connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43C	62-562	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-43C (cont)	62-550	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-549	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-546	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-579	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-555	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43C (cont)	62-570	No direct connection to RCS; Connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-574	No direct connection to RCS; Connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-43D	62-560	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-550	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-549	No direct connection to RCS; Normally operating system, continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore, release scenario is from containment atmosphere. Flow path is smaller in size than that required to result in a large release. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-43D (cont)	62-546	No direct connection to RCS; Normally operating system, continues to operate during accident. Therefore, not considered a path for release directly for RCS since flow continues to be forced into RCS; therefore release scenario is from containment atmosphere.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	62-576	Direct connection to the RCS. Normally operating system and continues to operate during accident, therefore not considered a path for release directly for RCS since flow continues to be forced into RCS. Therefore release scenario is from containment atmosphere. Open system IC to Closed system OC. Normally open valve IC.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-555	No direct connection to RCS; Normally closed vent/drain valve. Isolated from RCS by double check valves. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-568	No direct connection to RCS; Vent/Drain connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	62-572	No direct connection to RCS; Vent/Drain connected to containment atmosphere. Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-44	62-61	Direct connection to RCS; Normally operating system, penetration flow path connects open system IC to open system OC; normally open valve IC.	II-B-X44	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	62-639	Direct connection to RCS; Normally operating system, penetration flow path connects open system IC to open system OC; normally closed check valve IC.	II-B-X44	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-44 (cont)	62-63	Direct connection to RCS; Normally operating system, penetration flow path connects open system IC to open system OC; normally open valve OC.	II-B-X44	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-45	77-18	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-19	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-20	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	77-984	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-46	77-9	Normally operating system; RCS connection, however b/c of relief valve on RC drain tank and pump discharge pressure IC (SQN Dwg 47E8330-1), extremely unlikely to reach RCS pressure, therefore considered as containment atmosphere connection from open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-B-5	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	77-10	Normally operating system; RCS connection, however b/c of relief valve on RC drain tank and pump discharge pressure IC (SQN Dwg 47E8330-1), extremely unlikely to reach RCS pressure, therefore considered as containment atmosphere connection from open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an OC valve.	I-B-5	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-46 (cont)	84-511	Normally operating system; RCS connection, however b/c of relief valve on RC drain tank and pump discharge pressure IC (SQN Dwg 47E8330-1), extremely unlikely to reach RCS pressure, therefore given this scenario, flow path is also smaller in size than that required to result in a large release	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-47A	61-191	No connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	61-192	No connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	61-533	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-532	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-47B	61-193	No connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	61-194	No connection to RCS; penetration flow path connects open system IC to open system OC; normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	61-680	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	61-681	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-48A	72-547	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an IC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-39	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Same valve type. This is an OC valve. .	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-545	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-543	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-48B	72-548	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC. & OC. Different valve type. This is an IC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-2	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an OC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-546	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-544	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-49A	72-556	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an IC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	72-40 (FCV)	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an OC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-49A (cont)	72-215E	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-216E	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-215F	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-216F	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-40 (RFV)	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-552	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-49B	72-555	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an IC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-41 (FCV)	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC & OC. Different valve type. This is an OC valve.	I-B-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	72-217E	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	72-218E	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-217F	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-49B (cont)	72-218F	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-41 (RFV)	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	72-551	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-50A	70-87	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-687	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-90	No direct connection to RCS; penetration flow path connects open system IC to open system. OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-737	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-50B	70-679	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Different valve type. This is an IC valve.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	70-134	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Different valve type. This is an OC valve.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-678B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-51	26-1260	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally closed valve IC. NO valve OC. Different valve types. This is an IC valve.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	26-240	No direct connection to RCS; penetration flow path connects open system IC to closed system OC; normally open valve OC. Normally closed IC. Different valve types. This is an OC valve.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	26-1258	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-52	70-791	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-140	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-141	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-691B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-53	70-143	No direct connection to RCS; penetration flow path connects closed system IC to open system OC.	I-B-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-703	No direct connection to RCS; penetration flow path connects closed system IC to open system OC.	I-B-6	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	70-760	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-53 (cont)	70-702B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-765	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	70-702E	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-54	BLF	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Assume blind flanges to be normally closed valves. Same type.	I-A-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 72-hrs	168-hrs 168-hrs	168-hrs 72-hrs
X-56	67-1523D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-83	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-89	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve..	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	67-772	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-561D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-57	67-575D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-57 (cont)	67-111	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-112	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve..	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-58	67-1523A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-107	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-106	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-778	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-561A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-59	67-575A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-87	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve. .	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-88	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve. .	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-60	67-1523B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-90	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-91	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-774	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-561B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-61	67-575B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-103	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	67-104	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-62	67-1523C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-99	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-62 (cont)	67-105	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-776	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-561C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-63	67-575C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-95	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve IC & OC. Same valve type. This is an IC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	67-96	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC & IC. Same valve type. This is an OC valve.	I-A-3	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-64	31C-752	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	31C-223	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-222	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-65	31C-734	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-225	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-65 (cont)	31C-224	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-66	31C-715	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-230	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-229	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-67	31C-697	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-232	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	31C-231	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-68 Unit 2 Only	67-580D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-141	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-578D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-69 Unit 2 Only	67-580A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-130	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-69 Unit 2 Only (cont)	67-579A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-70 Unit 2 Only	67-585B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-297	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-139	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-71 Unit 2 Only	67-585C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-296	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	67-134	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-72 Unit 2 Only	67-585D	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-298	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-142	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-73 Unit 2 Only	67-585A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-73 Unit 2 Only (Cont)	67-295	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-131	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-74 Unit 2 Only	67-580B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-138	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-579B	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-75 Unit 2 Only	67-580C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	67-133	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	67-579C	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-76 Unit 1 Only	33-704	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	33-740	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	33-212	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-76 Unit 2 Only (cont)	33-722	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	33-739	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	33-211	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-77	59-633	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	59-522	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	59-529	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	59-704	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	59-651	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-78	26-1296	No direct connection to RCS; penetration flow path connects open system IC to open system OC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	26-243	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Normally open valve OC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	26-1293	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-79A	BLF	No direct connection to RCS; penetration flow path connects open system IC to open system OC. 1 flange IC and 1 flange OC used to isolate the penetration and analyze as normally closed valves of the same type.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
X-79B	BLF	No direct connection to RCS; penetration flow path connects open system IC to open system OC. 1 flange IC and 1 flange OC used to isolate the penetration and analyze as normally closed valves of the same type.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
X-80	30-40	No direct connection to RCS. This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Open system IC to open system OC. Normally closed valve IC & OC. Same valve type. This is an IC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	30-37	No direct connection to RCS. This valve is normally closed but is intermittently opened to provide for containment min-purge during power operation. Open system IC to open system OC. Normally closed valve OC & IC. Same valve type. This is an OC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-556TP	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-82	78-560	No direct connection to RCS; Penetration flow path connects open system IC to open system OC. Normally closed valve IC & OC. Same valve type. This is an IC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
	78-561	No direct connection to RCS; Penetration flow path connects open system IC to open system OC. Normally closed valve OC & IC. Same valve type. This is an OC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
	78-228A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-83	78-558	No direct connection to RCS; Penetration flow path connects open system IC to open system OC. Normally closed valve IC & OC. Same valve type. This is an IC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
	78-557	No direct connection to RCS; Penetration flow path connects open system IC to open system OC. Normally closed valve OC & IC. Same valve type. This is an OC valve.	I-A-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
	78-226A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-84A	68-308	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	68-307	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-85B		THE DP SENSORS ARE CLOSED SYSTEMS OUTSIDE OF CONTAINMENT THAT ARE ATTACHED DIRECTLY TO CONTAINMENT. NO ISOLATION VALVES ARE EMPLOYED FOR THESE SENSORS AS THEY USE A DOUBLE DIAPHRAGHM SYSTEM FOR DP MEASUREMENT. THE DIAPHGRAMS ARE QUALIFIED FOR POST-LOCA USE. NO DIRECT CONNECTION TO RCS. FLOW PATH SMALLER IN SIZE THAN THAT REQUIRED TO RESULT IN A LARGE RELEASE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-45Y	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-45X	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-85C	-	THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED. FLOW PATH SMALLER IN SIZE THAN THAT REQUIRED TO RESULT IN A LARGE RELEASE.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-86A	-	THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-86B	-	THIS LINE TRANSMITS PRESSURE FROM THE THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-86C	-	THIS LINE TRANSMITS PRESSURE FROM THE THIS LINE TRANSMITS PRESSURE FROM THE PRIMARY SYSTEM TO PRESSURE INSTRUMENTATION. THE LINE IS FLUID FILLED AND DOUBLE DIAPHRAGMED TO PREVENT COMMUNICATION BETWEEN THE PRIMARY SYSTEM FLUID AND THE AUXILIARY BUILDING. NO PRIMARY SYSTEM FLUID TRAVELS THROUGH THE PENETRATION SINCE THE INNER DIAPHRAGM IS LOCATED NEAR THE REACTOR VESSEL. SINCE DOUBLE DIAPHRAGMS ARE EMPLOYED FOR CONTAINMENT ISOLATION, NO CONTAINMENT ISOLATION VALVES ARE USED.	II,A#9	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-87B	52-502	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	52-503	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-87D	52-500	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	52-501	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-88	BLF	No direct connection to RCS; penetration flow path connects open system IC to open system OC. Assume normally closed valve IC and OC. Same valve type.	I-B-1	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-90 Unit 1 Only	32-287	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-80	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-285	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-281	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-90 Unit 2 Only	32-358	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-81	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-90 Unit 2 Only (cont)	32-353	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	32-354	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-91	43-251	Direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-250	Direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-497	Direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-92A, X-92B Unit 1 Only	43-207	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-452	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-424	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-208	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-453	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-423	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-92A, X-92B Unit 2 Only	43-207	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-210A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-525	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-417	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-208	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-210I	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-424	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-421	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-93	43-34	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-35	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-94A	90-109	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	90-107	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-94B	90-108	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	90-107	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-94C	90-110	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	90-111	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-95A	90-115	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	90-113	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-95B	90-114	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	90-113	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-95C	90-116	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	90-117	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-96C	43-22	Direct connection to RCS; Flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-23	Direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-97	30-134	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-135	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-98	52-506	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	52-507	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	52-508	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-99, X-100 Unit 1 Only	43-202	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-451	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-425	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-201	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	43-450	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-426	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-99, X-100 Unit 2 Only	43-202	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-200I	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-426	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-423	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-201	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-200A	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-427	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-419	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-101	43-319	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	43-318	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-474	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-102	3-352C	This line joins to the secondary side of the steam generator inside containment and is considered a closed system inside containment. Direct connection Closed system IC to open system OC. Normally closed valve OC.	III,A #1	System pressure boundary maintained System pressure boundary compromised	8 hrs 8 hrs	72 hrs 72 hrs	8 hrs 8 hrs
	Unit 2 Only 2-3-972	This valve is normally isolated from SG by valves 352A and 352B. Therefore flow is from containment atmosphere inside IC to OC. Open system IC to open system OC. Normally closed valve IC.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-103	43-461	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-317	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-341	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-464	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-104	3-351C	This line joins to the secondary side of the steam generator inside containment and is considered a closed system inside containment. Direct connection Closed system IC to open system OC. Normally closed valve OC.	III,A #1	System pressure boundary maintained System pressure boundary compromised	8 hrs 8 hrs	72 hrs 72 hrs	8 hrs 8 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	Unit 2 Only 2-3-970	This valve is normally isolated from SG by valves 352A and 352B. Therefore flow is from containment atmosphere inside IC to OC. Open system IC to open system OC. Normally closed valve IC.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-104 (cont)	Unit 2 Only 2-3-971	This valve is normally isolated from SG by valves 352A and 352B. Therefore flow is from containment atmosphere inside IC to OC. Open system IC to open system OC. Normally closed valve IC.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-106	43-460	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-325	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-307	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-469	No direct connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-107	74-2	RCS connection; standby system. Open system IC and OC. Normally closed valve IC downstream of another normally closed IC valve.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	74-1	RCS connection; standby system. Normally closed valve IC downstream of another normally closed IC valve.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
	74-505	Standby system, no direct RCS connection. Relief valve discharges to the pressurizer relief tank which does not reach RCS pressure. Therefore, flow path is smaller than minimum size required for a large release.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	74-504	No direct connection to RCS piping. Valve is IC. Only release path is from containment atmosphere to environment via the RHR system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-107 (cont)	74-503	No direct connection to RCS piping. Valve is IC. Only release path is from containment atmosphere to environment via the RHR system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	74-549	No direct connection to RCS piping. Valve is IC. Only release path is from containment atmosphere to environment via the RHR system. Flow path is smaller than minimum size required for a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-108	BLF	No connection to RCS; penetration flow path connects open system IC to open system OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
X-109	BLF	No connection to RCS; penetration flow path connects open system IC to open system OC.	Not Analyzed	System pressure boundary maintained System pressure boundary compromised	4-Hrs 4-Hrs	4-Hrs 4-Hrs	4-Hrs 4-Hrs
X-111	30-46	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	Pressure Boundary Maintained Pressure Boundary Compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-571	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	Pressure Boundary Maintained Pressure Boundary Compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
	30-46AX	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-46AY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-111 (cont)	30-46BY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-112	30-47	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	Pressure Boundary Maintained Pressure Boundary Compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-572	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	Pressure Boundary Maintained Pressure Boundary Compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-47AX	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-47AY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-47BY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-113	30-48	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
	30-573	No RCS connection. The containment vacuum relief isolation butterfly valve is located in series with the vacuum relief valve (spring loaded check valve) all outside of the containment. Open system IC and OC. Normally open valve OC.	I-A-4	System pressure boundary maintained System pressure boundary compromised	168-hrs 168-hrs	168-hrs 168-hrs	168-hrs 168-hrs
X-113 (cont)	30-48AX	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-48AY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	30-48BY	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-114	61-122	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-745	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-110	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-746	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs

SQN Containment Isolation Valve Completion Time Results							
SQN Pent #	**SQN Valve**	**Grouping Explanation**	**Group & Calc # (Note 1)**	**Maintenance Activity Type**	**ICLERP @ CT:**	**ΔLERF @ CT:**	**Justified CT**
X-115	61-97	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-692	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-96	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	61-691	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-116A	43-288	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-116A (cont)	43-287	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
	43-477	No connection to RCS; flow path smaller in size than that required to result in a large release.	Small Line	System pressure boundary maintained System pressure boundary compromised	168 hrs 168 hrs	168 hrs 168 hrs	168 hrs 168 hrs
X-117	BLF	No connection to RCS; penetration flow path connects open system IC to open system OC.	I-B-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs
X-118	BLF	No connection to RCS; penetration flow path connects open system IC to open system OC.	I-B-1	System pressure boundary maintained System pressure boundary compromised	72-hrs 72-hrs	168-hrs 168-hrs	72-hrs 72-hrs

Note 1 – Group/Calc # such as I,A#6, II,A#9, II,B#3, etc. match the generic configurations and use the CT times from the generic calculations for WCAP-15791-P-A, Revision 2. Group/Clac# such as I-A-1, I-B-3, II-B-2, etc. match the generic configurations of the WCAP but the CT times were determined by calculations using SQN specific PRA values. Group/Calc # III-A-12BC, III-A-13, III-A-14, II-A-X17, II-B-X44, II-A-BOUNDING do not match generic configurations of the WCAP and have been analyzed and CT times determined using SQN specific configurations and SQN specific PRA values. The evaluations for the CTs for all the CIVs are documented in TVAs PRA Evaluation Response, SQN-0-13-072. The CIVs marked as "Not Analyzed" either did not match the generic configurations and it was not advantageous to perform a specific SQN analysis that would increase the CT times greater than the original 4 hours or generic configurations did not yield a CT greater than the original 4 hours.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

Disposition of Existing License Amendment Requests

SII

DISPOSITION OF EXISTING LICENSE AMENDMENT REQUESTS

The following License Amendment Requests are under NRC review. The following table describes the request, and its affect on the ITS conversion, and its disposition.

DISPOSITION OF EXISTING LICENSE AMENDMENT REQUESTS				
Submittal Date	Description of Change	Affected ITS Submittal Sections/ Specifications	Affected CTS Pages	Disposition
August 10, 2012	Application to Revise Sequoyah Nuclear Plant Units 1 and 2 Updated Final Safety Analysis Report Regarding Changes to Hydrologic Analysis, (SQN-TS-12-02)	None	None	This is currently with the NRC for review.
January 7, 2013	Sequoyah Nuclear Plant, Units 1 and 2 License Renewal	None	None	This is currently with the NRC for review.
July 3, 2013	Application to Modify Ice Condenser Technical Specifications to Address Revisions in Westinghouse Mass and Energy Release Calculation (SQN-TS-12-04)	ITS: 3.6.12	Unit 1 3/4 6-26, 3/4 6-27 Unit 2 3/4 6-27, 3/4 6-28	Proposed changes are already reflected in this ITS submittal. Changes are annotated with an "A" DOC referencing the previously submitted LAR. See ITS 3.6.12 DOC A02.
October 2, 2013	Sequoyah Nuclear Plant (SQN), Units 1 and 2 - Proposed Technical Specification (TS) Change, "Ultimate Heat Sink (UHS) Temperature Limitations Supporting Alternate Essential Raw Cooling Water (ERCW) Loop Alignments (TS-SQN-13-01 and 13-02)"	~~ITS: 3.7.9~~ None	~~Units 1 and 2 3/4 7-14~~ None	~~Proposed changes are already reflected in this ITS submittal. Changes are annotated with an "A" DOC referencing the previously submitted LAR. See ITS 3.7.9 DOC A02.~~ This is currently with the NRC for review. Proposed changes are not reflected in the SQN ITS submittal. MEH-005

ENCLOSURE 8

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

Regulatory Commitments

REGULATORY COMMITMENTS

No.	Commitments for TSTF-411/418	Due Date/Event
1	Activities that degrade the availability of the AFW system, Reactor Coolant System (RCS) pressure relief system (pressurizer PORVs and safety valves), AMSAC, or Turbine Trip should not be scheduled when a logic cabinet is unavailable.	Upon Implementation
2	One complete Emergency Core Cooling System (ECCS) train that can be actuated automatically must be maintained when a logic cabinet is unavailable.	Upon Implementation
3	Activities that cause analog channels to be unavailable should not be scheduled when a logic cabinet is unavailable.	Upon Implementation
4	Activities on electrical systems (e.g., AC and DC power) and cooling systems (e.g., Essential Raw Cooling Water System (ERCW) and Component Cooling Water System (CCS) that support the systems or functions listed in the three commitments above (AFW, RCS pressure relief systems, AMSAC, Turbine Trip, or ECCS) should not be scheduled when a logic cabinet is unavailable. That is, one complete train of a function that supports a complete train of a function noted above must be available.	Upon Implementation
5	Activities that degrade the availability of the auxiliary feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a RTB is out of service.	Upon Implementation
6	Activities that degrade other components of the RPS, including master and slave relays, and activities that cause analog channels to be unavailable should not be scheduled when a RTB is unavailable.	Upon Implementation
No.	**Commitments for TSTF-427**	**Due Date/Event**
7	Sequoyah Unit 1 & Unit 2 will incorporate the guidance of NUMARC 93-01 Section 11, which provides guidance and details on the assessment and management of risk during maintenance.	Upon Implementation
8	Sequoyah Unit 1 & Unit 2 will revise procedures to ensure that the risk assessment and management process described in NEI 04-08 is used whenever a barrier is considered unavailable and the requirements of LCO 3.0.9 are to be applied, in accordance with an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified and avoided	Upon Implementation

No.	Commitments for TSTF-446	Due Date/Event	
9	Sequoyah Unit 1 & Unit 2 will implement the capability to assess the effect on incremental large early release probability when using the extended completion times for containment isolation valves in the program for managing risk in accordance with 10 CFR 50.65(a)(4) and the plant-specific configuration risk management program.	Upon Implementation	
No.	**Commitments for TSTF-493**	**Due Date/Event**	
10	Sequoyah will revise the UFSAR to include the methodologies used to determine the as-found and as-left tolerances for Limiting Safety Setting System (LSSS) instrument channel setpoints.	Upon Implementation	
11	Sequoyah will develop a monitoring program to adequately track the performance of Master Relays, Slave Relays, Logic Cabinets, Universal Logic Cards, Undervoltage Driver Cards, Safeguards Driver Cards, and Reactor Trip Breakers. (Reference Westinghouse Reports Section 3.2 and 3.5)	Upon Implementation	
No.	Commitment for ITS 3.7.12 Condition B	Due Date/Event	
12	Sequoyah will have guidance available describing compensatory measures to be taken in the event of an intentional or unintentional entry into ITS 3.7.12 Condition B.	Upon Implementation	MEH-006
No.	Commitment for ITS 3.9.4 Reviewer's Note	Due Date/Event	
13	The following guidelines are included in the assessment of systems removed from service during movement irradiated fuel: - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification OPERABILITY amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay. - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.	Upon Implementation	RPG-014

	The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."	

The above table identifies 13 ~~11~~ commitments by TVA in Enclosure 8 for the SQN conversion to Improved Technical Specifications license amendment request (LAR). Any other statements in this LAR submittal are provided for informational purposes and are not considered regulatory commitments. RPG-014

ENCLOSURE 9

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

GMW-003

List of ~~Required~~ Final Safety Analysis Report (FSAR) Descriptions For TSTF-500 and TSTF-400

LIST OF ~~REQUIRED~~ FINAL SAFETY ANALYSIS REPORT (FSAR) DESCRIPTIONS FOR TSTF-500 AND TSTF-400

GMW-003

The following table identifies FSAR descriptions for the Diesel Generator and Vital Batteries required by Sequoyah Nuclear Plant, Units 1 and Unit 2, as part of the adoption of TSTF-500, Revision 2. These changes will be included with the required implementation date in the Issuance of Amendment letter.

Additionally, an FSAR description for TSTF-400 is included.

REQUIRED FSAR DESCRIPTION	DUE DATE/EVENT
Sequoyah will change or verify that the FSAR: 1. Describes how a 5 percent design margin for the 125V Vital batteries corresponds to a 2 amp float current value indicating that the battery is 98 percent charged. 2. Describes how a 5 percent design margin for the Diesel Generator batteries corresponds to a 1 amp float current value indicating that the battery is 98 percent charged. 3. States that long term battery performance is supported by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 2.13 V per connected cell and that there are 60 connected cells in the battery, which corresponds to 127.8 V at the battery terminals. 4. Describes how the batteries are sized with correction margins that include temperature and aging and how these margins are maintained. 5. States the minimum established design limit for battery terminal float voltage. 6. States the minimum established design limit for electrolyte level. 7. States the minimum established design limit for electrolyte temperature. 8. Describes how each battery is designed with additional capacity above that required by the design duty cycles to allow for temperature variations and other factors. 9. Describes normal DC system operation (i.e., powered from the battery chargers) with the batteries floating on the system, and a loss of normal power to the battery charger describing how the DC load is automatically powered from the station batteries.	Upon implementation (applies to all)

10. Describes the availability of a means to charge the Vital Batteries and a description that the battery charger is capable of being supplied power from a power source that is independent of the offsite power supply. Specification 3.8.4, Required Action A.3	
11. Describes that DG tests verify that the critical protective trips that are not automatically bypassed perform their intended function.	Upon implementation

GMW-003

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

SQN Self-Identified Issues

Enclosure 2
SQN Self Identified Issues

During the NRC staff's review process, the staff had multiple requests for additional information (RAIs). In order to provide responses to the RAIs, SQN staff reviewed the SQN ITS conversion numerous times. As a result of the review, several issues were identified by SQN requiring revisions to the originally submitted ITS LAR. The table below provides information concerning justifications for the required revisions, ITS sections affected by the required revisions, and the location in the original LAR where the revisions occur.

Justification	Section	Page #
The Summary Disposition Matrix has been revised to indicate that CTS 3.9.3, Decay Time, is retained in ITS 3.9.8, Decay Time. This change ensures a fuel handling accident, involving recently irradiated fuel assemblies, does not occur.	Summary Disposition Matrix	Enclosure 2, Volume 1, page 30 of 37
CTS Section 2.1 Safety Limits, the Reactor Core Safety Limit Figure, 2.1-1 was originally proposed for relocation to the COLR and discussed in DOC LA01. This Figure will be retained in ITS 2.1.1 as Figure 2.1.1-1 and DOC LA01 will be deleted. This change maintains CTS.	ITS 2.1.1 (Units 1 & 2)	Enclosure 2, Volume 1, page 14 of 37 Enclosure 2, Volume 4, pages 5, 6, 8, 9, 11, 12, 14, 15a, 16, 17a, 22, 23, 30, and 31 of 38
Information has been added to the ISTS 3.0 Bases to clarify when SR 3.0.2 and SR 3.0.3 are applicable to Chapter 5 Specifications.	ITS 3.0 Bases	Enclosure 2, Volume 5, pages 57, 58, 59, 78, 79, 80, and 84 of 90
Inadvertent Omission of CTS SR number in DOC LA01 - Editorial	ITS 3.1.2 (DOC LA01)	Enclosure 2, Volume 6, page 44 of 356
Correction of inadvertent deletion of "INSERT 5" flag, JFD 4 indicators, and INSERT 5 heading and JFD 4 indicators on following page. - Editorial	ITS 3.1.7 Bases (Units 1 & 2)	Enclosure 2, Volume 6, pages 271, 272, 283 and 284 of 356
CTS 4.2.2.2.c.3 and CTS 4.2.2.2.c.4 require actions if the AFD min margin or the $f_2(\Delta I)$ min margin are < 0. Therefore, ITS SR 3.2.1.2 and SR 3.2.1.3 should verify the AFD min margin and the $f_2(\Delta I)$ min margin are ≥ 0. Associated changes will be required for JFD 4, JFD 6, and the ITS Bases. This change maintains CTS.	ITS 3.2.1 (Units 1 & 2)	Enclosure 2, Volume 7, pages 35, 37, 47, 49, 51, 70, and 93 of 249

Enclosure 2

SQN Self Identified Issues

CTS 4.2.3.2.c.3 and CTS 4.2.3.2.c.4 require actions if the $F_{\Delta H}$ min margin or the $f_1(\Delta I)$ min margin are < 0. Therefore, ITS SR 3.2.2.1 and SR 3.2.2.2 should verify the $F_{\Delta H}$ min margin and the $f_1(\Delta I)$ min margin are ≥ 0. Associated changes will be required for DOC M01, JFD 8, and the ITS 3.2.2 Bases. This change maintains CTS.	ITS 3.2.2 (Units 1 & 2)	Enclosure 2, Volume 7, pages 114, 126, 128, 129, 134, 136, 137, 138, 145, 149, 160, and 164 of 249
ITS 3.2.3 Bases correction associated with the details of the resolution from CSS-007. Resolution requires one additional sentence, "This ensures that the fuel cladding integrity is maintained for these postulated accidents," be deleted (see UFSAR 15.5.3). This change maintains current licensing basis.	ITS 3.2.3 Bases (Units 1 & 2)	Enclosure 2, Volume 7, pages 191 and 197 of 249
CTS Section 3.2, TABLE 3.2-1, DNB Parameters, contains the specific values for RCS average temperature, pressurizer pressure and RCS total flow rate. These specific values were proposed to be relocated to the COLR, as discussed in DOC LA01, and are now being retained in ITS LCO 3.4.1. Associated changes will be required for DOC LA03, DOC M01, ITS SRs 3.4.1.1, 3.4.1.2, 3.4.1.3, and 3.4.1.4, and the ITS 3.4.1 Bases. This change maintains CTS.	ITS 3.4.1 (Units 1 & 2)	Enclosure 2, Volume 9, pages 6, 7, 9, 10, 11, 11a, 13, 16, 18, 19, 20, 22, 23, 24, 28, and 37 of 696
Correction to Bases References section - Editorial	ITS 3.4.1 & ITS 3.4.12 (Units 1 & 2)	Enclosure 2, Volume 9, pages 34, 43, 421 and 441 of 696

Enclosure 2
SQN Self Identified Issues

CTS LCO 3.4.6.3 page used to markup the conversion to ITS SR 3.4.14.1 did not reflect the correct limit for RCS PIV leakage as compared to the controlled copy of the SQN TS. This change maintains CTS.	ITS SR 3.4.14.1 (Units 1 & 2)	Enclosure 2, Volume 9, pages 479, 484, 494, and 498 of 696
Correction of ITS SR 3.4.14.1 Surveillance Frequency to both the IST and SFCP. Associated changes will be required for the ITS 3.4.14 Bases. This change maintains CTS.	ITS SR 3.4.14.1 (Units 1 & 2)	Enclosure 2, Volume 9, pages 494, 498, 506, 508, 514 and 516 of 696
Correction of referenced ITS SR Number. - Editorial	ITS 3.4.14 - JFD 7	Enclosure 2, Volume 9, page 500 of 696
Correction of insert for Required Actions B.1 and B.2 - Editorial	ITS 3.4.14 Bases (Units 1 & 2)	Enclosure 2, Volume 9, pages 505 and 513 of 696
Correction of Required Action Header for B.1.1, B.1.2, and B.2 - Editorial	ITS 3.4.15 (Units 1 & 2) Bases	Enclosure 2, Volume 9, pages 560 and 568 of 696
Correction of Unit 2 specific information in "Insert 2" - This change maintains CTS.	ITS 4.0 (Unit 2)	Enclosure 2, Volume 15, page 29 of 35
Correction of acronym from ASME to ANSI for ANSI N510-1975. This change maintains CTS.	ITS 5.5.9 (Units 1 & 2)	Enclosure 2, Volume 16, pages 162 and 190 of 270
Restoration of the CTS acceptance criteria for P_a in ITS 5.5.14. This change maintains CTS.	ITS 5.5.14 (Unit 1 & 2)	Enclosure 2, Volume 16, pages 96, 127, 128, 140, 140a, 146, 147, 168, 169, 196, 197 and 205a of 270
CTS 6.9.1.14, COLR, information contained in CTS 2.1.1, 3.3.1, and 3.9.1 was proposed for relocation to the COLR. This information will be retained in ITS. Therefore, the list of proposed information relocated to the COLR will be revised. Associated changes are required for DOC A05. This change maintains CTS.	ITS 5.6.3 (Unit 1 & 2)	Enclosure 2, Volume 16, pages 213, 221, 227, 233 and 241 of 270
Correction in ISTS 5.7.2 - Editorial	ITS 5.7.2 (Unit 1 & 2)	Enclosure 2, Volume 16, page 263 and 267 of 270

Enclosure 2
SQN Self Identified Issues

Correction to status of TS-SQN-13-01 and 13-02 - Editorial	Enclosure 6	Enclosure 6, page E6-1