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See Proposed Changes to Tech Specs

SUBJECT: Forwards comments on draft improved TS SE & improved TS & Bases re proposed conversion from current TS to improved TS.

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March 27, 1998

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Reference: NRC letter dated February 25, 1998, from Jack N. Donohew, NRC, to James M. Levine, APS, "Draft Safety Evaluation Regarding Proposed Conversion to Improved Standard Technical Specifications for the Palo Verde Nuclear Generating Station, Unit No. 1 (TAC No. M96672), Unit No. 2 (M96673), and Unit No. 3 (M96674)

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Improved Technical Specifications – Comments on Draft SE and Submittal of ITS and Bases

In the referenced letter, the NRC staff provided to APS the draft Safety Evaluation (SE) related to the proposed conversion of the PVNGS Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS). The NRC staff requested comments on the draft SE, and a copy of the ITS, by March 17, 1998. In a telephone call on March 6, 1998, Mr. Jack Donohew of the NRC staff provided verbal approval to extend the submittal date until March 31, 1998, to provide additional time for a more thorough review of the SE.

APS is providing comments on the draft ITS SE, in Enclosure 1. A marked-up copy of the SE is provided in Enclosure 2.

In the referenced letter, the NRC staff also requested that APS submit a license condition for Appendix D to the licenses for PVNGS to ensure the timely relocation of CTS requirements associated with the implementation of the ITS. APS is proposing two license conditions in Enclosure 3. The first license condition would ensure that the requirements being relocated from the current technical

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specifications to licensee-controlled documents will be relocated prior to or upon implementation of ITS. The second license condition addresses the schedule for new and revised surveillance requirements. This provision will prevent situations where unnecessary and unwarranted surveillance testing would otherwise need to be performed immediately or shortly after ITS implementation. This provision was originally requested by APS in letter no. 102-03794-JML/AKK/SAB, dated October 4, 1996, and is modeled on provisions that have been granted by the NRC to other licensees implementing ITS.

The ITS is provided in Enclosure 4. The ITS Bases are provided in Enclosure 5.

Included in this submittal are two proposed changes to the ITS, two proposed changes to the ITS Bases, and several changes to the ITS justification packages, as discussed with the NRC staff. Enclosure 6 contains a summary of these changes. Enclosure 7 contains the replacement pages associated with these changes (the ITS and ITS Bases in Enclosures 4 and 5 have also been updated with these latest changes).

The ITS pages in Enclosure 4 are labeled as amendment 117, which is the next chronological amendment number for the PVNGS Unit 1 operating license. In order to enhance human performance factors, APS requests that the ITS be issued as amendment 117 for PVNGS Units 2 and 3 also. This request, which will require skipping several amendment numbers for Units 2 and 3, was discussed with Mr. Jack Donohew of the NRC staff.

There are no commitments being made or implied by this submittal. However, the following requirements and commitments are expected to be imposed with the issuance and implementation of the ITS and associated SE:

- The requirements of the proposed License Conditions in Enclosure 3;
- A commitment that PVNGS will add a listing to UFSAR Section 17.2 of the other UFSAR sections outside of Chapter 17 containing QA program commitments, as discussed on page 30 of the draft SE.
- A commitment that the PVNGS commercial-grade equipment certification program will detect the types of failures identified in the references cited in the SE, as discussed on page 56 of the draft SE.

The ITS implementation window for PVNGS is between July 1 and September 15, 1998, due to outage schedules. Based on this window, APS requests that the approval for the ITS amendment specify that the amendment be implemented by September 15, 1998.

Please contact Mr. Scott Bauer at (602) 393-5978 if you have any questions or would like additional information regarding this matter.

Sincerely,

JML/AKK/TNW/rlh

Enclosure 1: APS Comments on the Draft NRC Safety Evaluation for Proposed Conversion to Improved Technical Specifications for the Palo Verde Nuclear Generating Station
Enclosure 2: Marked-up Draft SE with APS Comments
Enclosure 3: Proposed License Conditions
Enclosure 4: ITS
Enclosure 5: ITS Bases
Enclosure 6: Summary of the ITS, ITS Bases, and ITS Justification Package Changes Included in This Submittal
Enclosure 7: Replacement Pages for ITS, ITS Bases, and ITS Justification Packages

cc: E. W. Merschoff (w/o enclosures)
K. E. Perkins (w/o enclosures)
M. B. Fields (w/o enclosures)
J. H. Moorman (w/o enclosures)
J. N. Donohew (with enclosures)
A. V. Godwin (with enclosures)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, J. M. Levine, represent that I am Senior Vice President - Nuclear, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

J. M. Levine

Sworn To Before Me This 27th Day Of March, 1998.

Notary Public

My Commission Expires

April 6, 1999

OFFICIAL SEAL
NORA E. MEADOR
Notary Public - State of Arizona
MARICOPA COUNTY
My Comm. Expires April 6, 1999

50-528 PALO VERDE 1 APSCO

APS COMMENTS ON THE DRAFT NRC SAFETY EVALUATION FOR PROPOSED CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS BOOK #1

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Enclosure 1

APS Comments on the Draft NRC Safety Evaluation for Proposed Conversion to Improved Technical Specifications for the Palo Verde Nuclear Generating Station

Enclosure 1

APS Comments on the Draft NRC Safety Evaluation for Proposed Conversion to Improved Technical Specifications for the Palo Verde Nuclear Generating Station

Page 1
In the first line, "*license*" should instead be "*licensee*" *(typo).*

Page 5
3.0 Evaluation
In the second paragraph, fifth line, add "NUREG Exception (NE)" to identify the terminology used by PVNGS for a justification for deviation from ISTS.

Page 8
In the second paragraph, "Categories *1*..." should instead be "Categories *I*..." (typo).

Page 8
In the line describing Category V, change "*Existing* LCO" to "*Exiting* LCOs," to correctly reflect the types of changes in Category V.

Page 10
In the Category V description, change "*Existing LCO*" to "*Exiting LCOs*" to correctly reflect that these are changes that affect the exiting from LCOs (going to a condition where the LCO would no longer apply).

Page 12
Section 1.0 – Less Restrictive
In paragraph L2, editorial suggestions are provided that would more clearly characterize the proposed/approved change.

Page 12
Paragraph L3 should be deleted because this proposed change was deleted as described in the "Summary of Enclosed ITS Changes" and the revised "Table of PVNGS Less Restrictive Changes (L)" in APS letter 102-04078, dated February 12, 1998.

Page 13
In paragraph L6, editorial suggestion to delete extra space.

Page 14
In paragraph L7, editorial suggestions are provided that would more clearly characterize the change to CTS.

Page 16
Section 3.1.1 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS.

Page 16
Section 3.1.2 – Less Restrictive
In paragraph L2, editorial suggestions are provided to more clearly characterize the change to CTS.

Page 16
Section 3.1.5 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS.

Page 17
Section 3.1.7 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS.

Page 17
Section 3.1.10 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS.

Page 18
Section 3.2.1 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS, and add the reference to TSTF 77 to clarify the source of the change.

Page 18
Section 3.2.4 – Less Restrictive
In paragraph L1, editorial suggestions are provided to more clearly characterize the change to CTS, and add the reference to TSTF 77 to clarify the source of the change.

Page 20
Section 3.3.5 – Less Restrictive
Paragraph L2 should be deleted. This proposed change is being re-categorized from a "unique" change to a "Category IX" change. (This re-categorization will make 3.3.5, L5 consistent with 3.3.2, L2.)

Page 21
Section 3.4.10 - Less Restrictive
In the first line of L1, "*ITS*" should instead be "*ISTS*," because the CTS is being revised to adopt the NUREG-1432 (ISTS) 3.4.10 provisions.
In the sixth line of L1, "*these* valves" should instead be "*two or more* valves," to correctly characterize that this change involves ITS 3.4.10 Action B which is two or more pressurizer safety valves inoperable.

Page 21
Section 3.5.2 – Less Restrictive
In the first line of LB1, "*reduction*" should instead be "*increase*," because changing the nitrogen cover pressure from 254 psig to 260 psig is an increase in nitrogen pressure and not a reduction.

Page 21
Section 3.6.3 – Less Restrictive
In the first and second lines of L8, insert the reference to TSTF-46, because it is the NRC-approved TSTF-46 revision to ISTS that is being adopted in this change to CTS 4.6.3.3.
In the sixth line of L8, change "values" to 'valves" (typo).

Page 22
Section 3.7.12 – Less Restrictive
In the first line of L2, "Control Room *Air Temperature Control System,"* should instead be "Control Room *Essential Filtration System"* because the latter is the actual title of CTS 3.7.7, which is being referenced.

Page 22 (and top of page 23)
Section 3.8.1 – Less Restrictive
In paragraph L2, editorial suggestions are provided that would more clearly characterize the proposed/approved change.

Page 23
Section 3.8.5 – Less Restrictive
In paragraph L2, change "IST" to "ITS" (typo).

Page 23
Section 3.8.5 – Less Restrictive
In paragraph L3, delete reference to Action A.3 of ITS 3.8.5, because this action does not exist.

Page 24
Section 3.8.8 – Less Restrictive
In paragraph L2, editorial suggestions are provided that would more clearly characterize the proposed/approved change.

Page 24
Section 3.8.10 – Less Restrictive
In paragraph L3, editorial suggestions are provided that would more clearly characterize the proposed/approved change.

Page 25
In the first line at the top of the page, the words "instead of *referring to* the actions *are required...*," should instead be "instead of *requiring* the actions...," in order to more accurately characterize the change.

Page 25
Section 5.0 – Less Restrictive, L1
In the first line, "Department Leader" should instead be "Plant Superintendent (identified at PVNGS as Department Leader, Operations)," because ISTS 5.1.1 uses the title Plant Superintendent.

Page 25
Section 5.0 – Less Restrictive, LB1
In the sixth line, "this distance is updated to *metric units*" should instead be "this distance is updated to *30 centimeters*," because this change is revising the radiation measurement distance from 18 inches to 30 centimeters (to be consistent with 10 CFR 20), and not just changing the units to metric.

Page 27
Indent the first paragraph to be consistent with the format to be carried over from the previous page.

Page 27
Descriptions of System Operation (Type 2)
Add a reference to the controls specified in 10 CFR 50.54(a) for changes to the QA program description, because some of the Type 2 changes are relocations to the QA program description.

Page 27
Descriptions of System Operation (Type 3)
Add a reference to the QA program description, because some of the Type 3 changes are relocations to the QA program description.

Page 28
In the next to last paragraph, add the other licensee controlled documents that will contain relocated CTS requirements, as listed in table LA, so that this discussion is consistent with table LA.

Page 29
Relocations to the Quality Assurance Program
In the first and second paragraphs, a revision is suggested to correctly characterize that the November 26, 1997 letter from APS to NRC did not request changes. The November 26 letter submitted revised pages to the quality assurance program description in the UFSAR that added commitments replicating the CTS QA related requirements proposed for relocation. The QA program description changes submitted in the November 26, 1997 letter have been implemented without prior NRC approval under the provisions of 10 CFR 50.54(a). The QA program description changes were implemented and submitted to provide the NRC staff additional confidence that the proposed CTS QA related relocations would be properly relocated prior to their approval of ITS.

Page 29
Relocations to the Quality Assurance Program
In the first paragraph under this section, it is suggested that a table number or name be specified for the "table beginning on page 32" to be consistent with approach of designating other tables in the SE.

Page 29
Relocations to the Quality Assurance Program
In the first paragraph, the fourth line, a revision is suggested to more correctly characterize that the three details identified are not all inclusive of the proposed QA related relocations. There are also QA related relocations included in the table beginning on page 32 that would be categorized as "other changes" as specified in GL 95-06.

Page 29
Relocations to the Quality Assurance Program
In the last paragraph, a revision is suggested to make the sentences more complete.

Page 30
In the first paragraph, a revision is suggested to more directly identify that the APS letter dated February 12, 1998 contained the most current version of the table of proposed relocated CTS QA requirements. The tables of the proposed QA relocations in the December 16, 1997 letter are superseded and replaced by the February 12, 1998 letter, and the November 26, 1997 letter did not request changes.

Page 30
In the first paragraph, a revision is suggested to delete the specific CTS sections. The sections identified in this paragraph are not all inclusive of the CTS QA requirements to be relocated. Since the CTS QA relocations are fully identified in the table beginning on page 32, referencing the table would eliminate any redundancy and potential inaccuracy of including only some of the relocations in this paragraph. **If there is a need to identify the category of each specific CTS section change, a "category" column could be added to the table.** Also, the reference to the three CTS QA categories to be relocated is redundant to the same information already specified on page 29.

Page 30
In the second paragraph, fourth line, "the licensee *added* a listing to UFSAR section 17.2 of the other UFSAR sections *contained in the* QA program outside of UFSAR chapter 17 *as follows*" should instead be "the licensee *will add* a listing to UFSAR Section 17.2 of the other UFSAR sections *outside of UFSAR Chapter 17 that contain* QA program *commitments*." The listing has not yet been added to the UFSAR, and this suggested rewording more clearly characterizes that there are QA program commitments that are in other sections of the UFSAR Chapter 17 that are to be controlled under 10 CFR 50.54(a).

Page 30
Delete the third paragraph. This paragraph is redundant to the second paragraph.

Page 30
In the bulleted list below the third paragraph, change "requirements" to "commitments." The CTS QA related requirements become commitments when they are relocated to the QA program description and subject to the change controls of 10 CFR 50.54(a). The provisions of 10 CFR 50.54(a) specify that NRC approval is required prior to making changes to the QA program description that reduce the *commitments* previously accepted by the NRC.

Page 31
In the first paragraph, fourth line, editorial suggestion.

Page 31
Near the bottom of the second paragraph, change "IST" to "ITS" (typo).

Page 32
Table - Relocations to the QA Program
CTS 6.2.2.e
The line containing CTS 6.2.2.e should be deleted. CTS 6.2.2.e is being relocated to the UFSAR (but not the QA program description), as described in letters 102-03942, dated May 30, 1997, 102-03972, dated July 18, 1997, and 102-04078, dated February 12, 1998. Fire team manning requirements, as well as shift manning requirements, are not specifically designated as relocated to the QA Program Description. These requirements are, however, being relocated to a section of the UFSAR that also contains QA program commitments.

Page 32
Table - Relocations to the QA Program
CTS 6.5.1.6
The comments for CTS 6.5.1.6 should be deleted. Items 13.4.2.6.g. and h. were not new responsibilities added in items g and h, but were relocated from CTS Table 3.3-1 Action 2, CTS Table 3.3-3 Action 13, CTS 6.13.b, CTS 6.14.b, and the BASES for CTS 3/4.7.9 (as described later in the SER). This comment is not necessary.

Page 33
Table - Relocations to the QA Program
CTS 6.5.3.4
In the CTS section and UFSAR section columns, (*a-l*) should instead be (*a-i*).

Page 34
Table - Relocations to the QA Program
CTS 6.8.1.j
The Program column should include "+ other," because some of the relocated requirements are outside of QA program commitments.

Page 34
Table - Relocations to the QA Program
CTS 6.8.1.n
In the Comments column, change "are" to "is" to match the singular procedural requirement being relocated.

Page 35
Table - Relocations to the QA Program
CTS 6.8.4
In the Comments column, add "Programs are retained in the ITS, described in UFSAR 13.5, or contained in other licensee-controlled documents. Audit requirements are relocated to UFSAR 13.4.5 as QA program commitments."
These comments clarify that the audit requirements are the relocated requirements that are QA program commitments.
In the Program column, add "+ other," to clarify that some of the CTS requirements being relocated are outside of QA program commitments.

Page 35
Table - Relocations to the QA Program
CTS 6.8.4.b., d., and f.
Delete the lines for CTS 6.8.4.b(program), d(program), and f(program). These CTS program requirements are being relocated to the UFSAR (but not the QA program description), as described in letters 102-03942, dated May 30, 1997, 102-03972, dated July 18, 1997, and 102-04078, dated February 12, 1998.

Page 35
Table - Relocations to the QA Program
CTS 6.8.4.b and d.
In the comment sections for CTS 6.8.4.b and d, add "Audit requirement only," to be consistent with the other table entries.

Page 35
Table - Relocations to the QA Program
CTS 6.8.4.f.
In the comment column for CTS 6.8.4.f, an editorial revision is suggested to more clearly characterize the relocation.

Page 36
Table - Relocations to the QA Program
CTS 6.10.2.i
Suggestions are provided in UFSAR column of CTS 6.10.2.i to more accurately identify the UFSAR sections that will contain the relocated QA program commitments. Suggestions are provided in the Comments column to more accurately define how the CTS requirement is being relocated as a QA program commitment.

Page 37
Table - Relocations to the QA Program
CTS 6.11/6/11/1
Suggestion is provided in the Comments column to more clearly define the relocation.

Page 38
CTS 3.1.2.1, Flow Paths – Shutdown
In the first line, add a reference to CTS 4.1.2.1 to explicitly identify this relocation and be consistent with other entries in this section.

Page 39
In the second paragraph, first line, editorial changes are provided for clarification.

Page 39
CTS 3.1.2.2, Flow Paths – Operating
In the first paragraph, first line, add a reference to CTS 4.1.2.2.1 and 4.1.2.2.2 to explicitly identify this relocation and be consistent with other entries in this section.
In the fifth line, editorial changes are provided for clarification.

Page 39
CTS 3.1.2.3, Charging Pumps – Shutdown
In the first paragraph, first line, add a reference to CTS 4.1.2.3 and 4.1.2.1 to explicitly identify this relocation and be consistent with other entries in this section.

Page 40
In the second paragraph, first line, editorial changes are provided for clarification.

Page 40
CTS 3.1.2.4, Charging Pumps – Operating
In the first paragraph, first line, add a reference to CTS 4.1.2.4 to explicitly identify this relocation and be consistent with other entries in this section.
In the third line, editorial changes are provided for clarification.

Page 41
CTS 3.1.2.5, Borated Water Sources – Shutdown
In the first paragraph, first line, add a reference to CTS 4.1.2.5 to explicitly identify this relocation and be consistent with other entries in this section. In the third and fifth lines, change "and" to "or" to properly characterize this CTS requirement that only *one* of the two water sources must be operable.
In the fourth line, add "each" after "gallons" to properly characterize the CTS requirement that *each* water source contain at least 33,500 gallons.
In the second paragraph, first line, editorial changes are provided for clarification.

Page 41
CTS 3.1.2.6, Borated Water Sources – Operating
In the first paragraph, first line, and second paragraph, ninth line, add a reference to CTS 4.1.2.6 to explicitly identify this relocation and be consistent with other entries in this section. In the first paragraph, fourth line, change "33,500 gallons" to "as specified in Figure 3.1-1," to correctly reflect the requirements in CTS 3.1.2.6. In the first and second paragraphs, additional editorial changes are provided for clarification, and to specify that only the SFP requirements are being relocated (the CTS 3.1.2.6 RWT requirements are being retained in the ITS, as described in the Split Report).

Page 42
CTS 3.3.1, (Table 3.3-1, Item D), Supplementary Protective System
Change the title of this section from "protective system" to "protection system" to correctly reflect the CTS section title.
In the third line, add "supplementary protection system pressurizer" to properly identify the parameter being relocated.

Page 42
CTS 3.3.3.1, (Table 3.3-6, Item 1.A), Fuel Pool Area Monitor (RU-31)
In the fourth, fifth and tenth lines, revise wording to clarify that the radiation monitor alarms at specified limits. The CTS 3.3.3.1 Fuel Pool Area Monitor requirements do not impose radiation limits on the area or ensure that a radiation level has not been exceeded. The CTS only require an instrument to monitor radiation and alarm/trip at a preset level.

Page 42
CTS 3.3.3.1, (Table 3.3-6, Item 1.B), New Fuel Area Monitor (RU-19)
In the fourth, fifth and seventh lines, revise wording to clarify that the radiation monitor alarms at specified limits. The CTS 3.3.3.1 New Fuel Area Monitor requirements do not impose radiation limits on the area or ensure that a radiation level has not been exceeded. The CTS only require an instrument to monitor radiation and alarm/trip at a preset level.

Page 43
CTS 3.3.3.1, (Table 3.3-6, Item 1.E), Main Steam Area Monitors (RU-139 A&B and RU-140 A&B)
In the fifth and eighth lines, revise wording to clarify that the radiation monitor alarms at specified limits. The CTS 3.3.3.1 Main Steam Area Monitors requirements do not impose radiation limits on the area or ensure that a radiation level has not been exceeded. The CTS only require an instrument to monitor radiation and alarm/trip at a preset level.

Page 43
CTS 3.3.3.2, Incore Detectors
In the first line, add a reference to CTS 3.3.3.2 and 4.3.3.2 to explicitly identify these relocations and be consistent with other entries in this section.

Page 44
CTS 3.3.3.3, (Table 3.3-7), Seismic Monitoring
In the first and second lines, add a reference to CTS 3.3.3.3, 4.3.3.3.1 and 4.3.3.3.2 to explicitly identify these relocations and be consistent with other entries in this section, and make editorial changes to eliminate redundancy.

Page 44
CTS 3.3.3.4, (Table 3.3-8), Meteorological Instrumentation
In the first and second lines, add a reference to CTS 3.3.3.4, 4.3.3.4 and Table 4.3-5 to explicitly identify these relocations and be consistent with other entries in this section

Page 44
CTS 3.3.3.6, Post-Accident Monitoring Instrumentation
In the first line, add a reference to CTS 3.3.3.6 and 4.3.3.6 to explicitly identify this relocation and be consistent with other entries in this section.

Page 45
In the first paragraph, in the fifth line, revise wording to properly characterize that the May 9, 1988, NRC letter to CEOG referred to each plant's SE, and did not specifically refer to the PVNGS SE. Also, delete an incorrect reference to a March 5, 1987, PVNGS SE.
In the sixth line, add a sentence to refer to the correct June 18, 1985, SE that documented the NRC approval of the PVNGS conformance with RG 1.97.

Page 45
CTS 3.3.3.7, (Table 3.3-11), Loose-Parts Detection Instrumentation
In the first line, add a reference to CTS 4.3.3.7 to explicitly identify this relocation and be consistent with other entries in this section.

Page 46
CTS 3.3.3.8, (Table 3.3-12), Explosive Gas Monitoring Instrumentation
In the first line, add a reference to CTS 4.3.3.8 to explicitly identify this relocation and be consistent with other entries in this section.

Page 46
CTS 3.4.3.2, Auxiliary Spray
In the first line, add a reference to CTS 3.4.3.2, 4.4.3.2.1, 4.4.3.2.2, and 4.4.3.2.3 to explicitly identify this relocation and be consistent with other entries in this section.

Page 46
CTS 3.4.6, (Table 3.4-2), RCS Chemistry
In the first line, add a reference to CTS 4.4.6 to explicitly identify this relocation and be consistent with other entries in this section.

Page 47
CTS 3.4.8.2, Pressurizer Heatup/Cooldown Limits
In the second line, add a reference to CTS 4.4.8.2.1 and 4.4.8.2.2 to explicitly identify this relocation and be consistent with other entries in this section.
In the seventh line, make an editorial change to eliminate redundancy.

Page 47
CTS 3.4.10, Reactor Coolant System Vents
In the first line, add a reference to CTS 4.4.10 to explicitly identify this relocation and be consistent with other entries in this section.

Page 48
In the first paragraph, second line, change "TS" to "ITS" for consistency.

Page 48
CTS 3.6.4.3, Hydrogen Purge Cleanup System
In the fifth line, "operable" should be changed to "inoperable," to correctly reflect the CTS requirement.

Page 49
CTS 3.7.9, Snubbers
In the ninth line, "valves" should be changed to "snubbers," to properly reflect the proposed change.

Page 52
In the first paragraph, second line, "cane" should be changed to "crane" (typo).

Page 52
CTS 3.7.12, Fuel Building Essential Ventilation
Editorial suggestions are provided that would more clearly characterize the proposed/approved change.

Page 52
CTS 3.10.6, Special Test Exception – Safety Injection Tanks
Correct "SR 4.6.10.1" to "SR 4.10.6.1." (typo)

Page 54
F. Control of Specifications, Requirements, and Information Removed from CTS.
In the first paragraph under this section, fifth line, "record" should instead be "records" (typo).

Page 54
F. Control of Specifications, Requirements, and Information Removed from CTS.
In the second paragraph under this section, first line, a revision is suggested to more directly identify that the February 12, 1998 letter, with revisions in the March XX, 1998 letter [this submittal], contained the most current version of the matrix of proposed relocated CTS requirements. The tables of the proposed CTS relocations in the May 30 and December 16 1997 letters were superseded and replaced by those in the February 12, 1998, and March XX, 1998, letters.

Page 54
F. Control of Specifications, Requirements, and Information Removed from CTS.
In the second paragraph under this section, third line, "IST" should instead be "ITS" (typo).

Page 54
F. Control of Specifications, Requirements, and Information Removed from CTS.
In the second paragraph under this section, beginning with the fourth line, editorial suggestions are provided to make this section consistent with the proposed license condition that will require APS to relocate the CTS requirements identified in the matrix.

Page 55
In the first paragraph, editorial suggestions are provided to more clearly identify the proper interpretation of the intent of theses requirements. It appears that the intent of these requirements are for the licensee to maintain an auditable record of, and establish an implementation schedule for, procedure changes (not to maintain an implementation schedule for procedure changes) associated with the implementation of (not development of) ITS.

Page 55
Recommend deleting the second paragraph because it is redundant with the description of the relocation license condition provided on page 54.

Page 56
In the last paragraph, last line, add "in ESFAS applications." This SE discussion supports the proposed changes to ESFAS requirements.

Page 57
ITS 3.5.2, CTS LCO 3.5.1
In the third paragraph under this section, the second, third, and fourth lines, the references to minimum and maximum borated SIT water volumes that are expressed in cubic feet should instead be expressed in percent wide range indication. CTS LCO 3.5.1 identifies the SIT water volumes in both percent wide range indication and in cubic feet; however, only the CTS percent wide range indications are being carried over to ITS. The SIT cubic feet requirements are being relocated to the Bases and updated under 10 CFR 50.59.

Page 59
ITS 3.6.6, CTS LCO 3.6.2.1
The changes identified for this section will more accurately characterize and justify the proposed CTS change. In addition, the wording changes will eliminate incorrect, outdated PVNGS design details that are not contained in the current UFSAR.

Page 60
ITS 3.6.6, CTS SR 4.6.2.1.c
In the second paragraph under this section, the fifth and sixth lines, the parenthetical statement "i.e., 32 seconds for PVNGS per Combustion Engineering Safety Analysis Report (CESSAR) System 80, Appendix 6A, paragraph 6.3" should instead be "i.e., 90 seconds for PVNGS per Updated Final Safety Analysis Report (UFSAR) section 6.5.2.8, paragraph (RA) 7.10.4." The PVNGS UFSAR explicitly takes an exception to the containment spray flow delivery time given in CESSAR Appendix 6A, as described in UFSAR 1.9.2.4.1 and 6.5.2.8.

Page 61
ITS 5.0, CTS SR 4.6.4.3.d.1
This entire discussion should be deleted, since the proposed change is no longer part of the ITS conversion. Although the hydrogen purge system was initially designated to be in the ITS, it was re-designated to be relocated to the TRM as described in the letter to NRC dated September 18, 1997, RAI response number 3.6.7-1. The relocation of the hydrogen purge system requirements of CTS 3.6.4.3 is also documented in the "R" table attached with the SE. The revision to the allowable pressure drop across the hydrogen purge exhaust air filtration unit will be controlled under 10 CFR 50.59, consistent with its relocation to the TRM.

Pages 63 and 64
ITS 3.8.2, CTS LCO 3/4.8.2, "Electrical Power Systems – DC Sources," replacement of existing round cells with low specific gravity cells manufactured by vendors not specified in CTS.
Recommend deleting this discussion because the subject proposed CTS changes were approved by the NRC as amendments 116, 109, and 88 to Units 1, 2, and 3, respectively, on March 16, 1998. Any redundant discussion in this SE could introduce inconsistenties with the SE for the approved amendments, and could confuse readers as to the purpose of including this SE discussion even though it was previously approved.

Table of PVNGS Administrative Changes (A)
No comments or changes to this table.

Table of PVNGS More Restrictive Changes (M)
Page 15
3.8.1 M.2
For DOC 3.8.1 M.2, the CTS reference should be changed to LCO 3.8.1.1 to correct an incorrect SR reference.

Table of PVNGS More Restrictive Changes (M)
Page 15
3.8.1 M.3
In the first line for DOC 3.8.1 M.3, the ITS reference should be changed to SR 3.8.1.12.a to correct an incorrect SR reference.

Table of PVNGS More Restrictive Changes (M)
Page 15
3.8.3 M.1
In the first line for DOC 3.8.3 M.1, the CTS reference should be changed to LCO 3.8.1.3.1 and the ITS reference should be changed to LCO 3.8.3 Action B to correct incorrect references.

Table of PVNGS More Restrictive Changes (M)
Page 16
3.8.3 M.1
In the first entry on this page for DOC 3.8.3 M.1, the CTS reference should be changed to LCO 3.8.1.1 and LCO 3.8.1.2 and the ITS reference should be changed to LCO 3.8.3 Action E to correct incorrect references.

Table of PVNGS More Restrictive Changes (M)
Page 16
3.8.3 M.1
In the second entry on this page for DOC 3.8.3 M.1, the CTS reference should be changed to LCO 3.8.1.1 and LCO 3.8.1.2 to correct incorrect references.

Table of PVNGS More Restrictive Changes (M)
Page 16
3.8.3 M.1
In the third entry on this page for DOC 3.8.3 M.1, the CTS reference should be changed to LCO 3.8.1.1 and LCO 3.8.1.2 to correct incorrect references.

Table of PVNGS More Restrictive Changes (M)
Page 16
3.8.3 M.3
In the entry for DOC 3.8.3 M.3, the CTS reference should be changed to SR 4.8.1.3.1.1 and SR 4.8.1.3.1.2 to correct incorrect references.

Table of PVNGS Relocated Details (LA),
Page 3
3.1.7
For ITS 3.1.7, LA.1 for CTS 3.1.3.6, the Destination Document should changed to the COLR instead of UFSAR. This relocation is addressed in the Summary of Changes included with this submittal.

Table of PVNGS Relocated Details (LA),
Page 29
Add the Core Operating Limits Report (COLR), controlled by ITS 5.6 and 10 CFR 50.59, to reflect the relocation LA.1 for CTS 3.1.3.6 (LCO Note ##) (ITS 3.1.7). This relocation is addressed in the Summary of Changes included with this submittal.

Table of PVNGS Less Restrictive Changes (L)
Page 11
3.3.5 L.2
For DOC 3.3.5 L.1, the Category should be changed from "unique" to Category IX. This change is addressed in the Summary of Changes included with this submittal.

Table of PVNGS Relocated Specifications (R)
No comments or changes to this table.

Enclosure 2

Marked-up Draft Safety Evaluation with APS Comments

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-41, AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-51, AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-74 ARIZONA PUBLIC SERVICE COMPANY, ET AL. PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3 DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

Arizona Public Service Company (the licensee) has been operating Palo Verde Nuclear Generating Station (PVNGS) with Technical Specifications (TS) which were issued with the original operating license on December 31, 1984, December 9, 1985, and March 25, 1987, for Units 1, 2 and 3, respectively, as amended from time to time. In its application dated October 4, 1996, as supplemented by (1) 19 letters in 1997 dated: January 31, March 16, May 30 (2 letters), June 6, July 18 (5 letters), August 31, September 18 (2 letters), September 19 (2 letters), November 7, November 14, November 26, and December 16, and (2) the letter of February 12, 1998, the licensee proposed to convert the current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) for PVNGS. The conversion is based on NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 1, dated April 1995, and on guidance provided in the Nuclear Regulatory Commission's (NRC's) "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published on July 22, 1993 (58 FR 39132).

The licensee submitted this request on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority. The overall objective of the proposed amendments are to, consistent with the Commission's Final Policy Statement, rewrite, reformat, and streamline the TS for PVNGS to be in accordance with 10 CFR 50.36, "Technical Specifications."

Hereafter, the proposed Technical Specifications are referred to as the ITS, the existing Technical Specifications are referred to as the CTS, and the Technical Specifications in

NUREG-1432 are referred to as the Improved Standard Technical Specifications (ISTS). The corresponding Bases to Technical Specifications are the ITS Bases, CTS Bases, and ISTS Bases, respectively.

In addition to basing its ITS on the ISTS and the Commission's Final Policy Statement, the licensee retained portions of the CTS as a basis for the ITS. Plant-specific issues, including design features, requirements, and operating practices, were discussed with the licensee during the meeting on June 26, 1996 (summary issued on July 16, 1996). Based on these discussions, the licensee submitted its application dated October 4, 1996. In addition, the licensee proposed matters of a generic nature that were not in the ISTS. The NRC staff requested that the licensee submit such generic issues as a proposed change to ISTS through the Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These generic issues were considered for specific applications in the PVNGS ITS. Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents (i.e., documents, such as the Updated Final Safety Analysis Report (UFSAR) for an operating plant, for which changes to the documents by licensees are controlled by a regulation such as 10 CFR 50.59), whereas NRC-controlled documents, such as the TS, may not be changed by the licensee without prior staff approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The Commission's proposed action on the PVNGS application for amendments dated October 4, 1996, and supplemented by letter dated March 16, 1997, was published in the FEDERAL REGISTER on April 14, 1997, (62 FR 18153). The NRC staff's evaluation of the application, the letter of March 16, 1997, and the additional supplements to the licensee's application listed above that resulted from NRC requests for information and discussions with the licensee during the NRC staff review, is presented in this Safety Evaluation. These plant-specific changes serve to clarify the ITS with respect to the guidance in the Final Policy Statement and ISTS. Therefore, the changes are within the scope of the action described in the FEDERAL REGISTER notice.

During its review, the NRC staff relied on the Final Policy Statement and on the ISTS as guidance for acceptance of CTS changes. This Safety Evaluation provides a summary basis for the NRC staff conclusion that the licensee can develop ITS based on the ISTS, as modified by plant-specific changes, and that the use of the ITS is acceptable for continued operation. The NRC staff also acknowledges that, as indicated in the Final Policy Statement, the conversion to the ITS is a voluntary process. Therefore, it is acceptable that the ITS differs from ISTS, reflecting the current licensing basis for PVNGS. The NRC staff approves the licensee's changes to the CTS with modifications documented in the revised submittals.

For the reasons stated *infra* in this Safety Evaluation (SE), the NRC staff finds that the ITS issued with this license amendment comply with Section 182a of the Atomic Energy Act, 10

CFR 50.36, and the guidance in the Final Policy Statement, and that they are in accord with the common defense and security and provide adequate protection of the health and safety of the public.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [Such technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968). Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SR); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility Owners Groups and the NRC staff developed ISTS that would establish models of the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and Owners Groups developed generic administrative and editorial guidelines in the form of a "Writers Guide" for preparing technical specifications, which gives greater consideration to human factors principles and was used throughout the development of licensee-specific ITS.

In September 1992, the Commission issued NUREG-1432 for Combustion Engineering plant ISTS, which was developed using the guidance and criteria contained in the Commission's interim policy statement. ISTS were established as a model for developing improved TS for

Combustion Engineering plants in general. ISTS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the Nuclear Steam System Supplier (NSSS) Owners Groups in May 1988. ISTS also reflect the results of extensive discussions concerning various drafts of ISTS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1432 provide an abundance of information regarding the extent to which the ISTS presents requirements that are necessary to protect the public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the ISTS, and encouraged licensees to use the ISTS as the basis for plant-specific TS amendments, and for complete conversions to ISTS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated surveillances should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [There is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 *FR* 36593, July 19, 1995). The Final Policy Statement criteria 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.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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.

Criterion 4

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

Part 3.0 of this Safety Evaluation explains the NRC staff's conclusion that the conversion of the PVNGS CTS to those based on ISTS, as modified by plant-specific changes, is consistent with the PVNGS current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

3.0 EVALUATION

The NRC staff's ITS review evaluates changes to the CTS that fall into five categories defined by the licensee and includes reviewing whether existing regulatory requirements are adequate for controlling future changes to requirements removed from the CTS and placed in licensee-controlled documents. This evaluation also discusses the NRC staff's plans for monitoring the licensee's implementation of these controls at PVNGS.

NUREG exception (NE) or

In addition to the initial submittal of October 4, 1996, as supplemented by 20 letters, the staff review identified the need for clarifications and additions to the submittal in order to establish an appropriate regulatory basis for translation of CTS requirements into the ITS. Each change proposed by the licensee in the amendment request is identified as either a discussion of change (DOC) to the CTS or a justification for deviation (JFD) from the ISTS. The NRC staff comments were documented as requests for additional information (RAIs) and forwarded to the licensee for response by 11 letters in 1997 dated: January 16, April 8, April 10, April 22, April 25, May 16, and June 6 (5 letters). The licensee provided written responses to the staff requests in 16 letters in 1997 dated March 16, May 30, June 6, July 18 (5 letters), August 31, September 18 (2 letters), September 19 (2 letters), November 7, November 14, and December 16, and in the letter of February 12, 1998. The docketed letters clarified and revised the licensee basis for translating CTS requirements into the ITS. The NRC staff finds the licensee's submittals provided sufficient detail to allow the staff to reach a conclusion regarding the adequacy of the licensee's proposed changes to the CTS to create the ITS for PVNGS.

The license amendment application was organized such that changes were included in each of the following CTS change categories, as appropriate:

(1) Administrative Changes, (A), i.e., non-technical changes in the presentation of existing requirements;

(2) Technical Changes - More Restrictive, (M), i.e., new or additional CTS requirements;

(3) Technical Changes - Less Restrictive (specific), (L), i.e., changes, deletions and relaxations of existing TS requirements;

(4) Technical Changes - Less Restrictive (generic), (LA), i.e., deletion of existing TS requirements by movement of information and requirements from existing specifications (that are otherwise being retained) to licensee-controlled documents, including TS Bases; and

(5) Relocated Specifications, (R1), i.e., relaxations in which whole specifications (the LCOs and associated action and SR) are removed from the CTS (an NRC-controlled document) and placed in licensee-controlled documents.

These general categories of changes to the licensee's CTS requirements and ISTS differences may be better understood as follows:

A. Administrative Changes

Administrative (non-technical) changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used ISTS as guidance to reformat and make other administrative changes. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

(1) providing the appropriate numbers, etc., for ISTS bracketed information (information that must be supplied on a plant-specific basis and that may change from plant to plant)

(2) identifying plant-specific wording for system names, etc.

(3) changing the wording of specification titles in the ISTS to conform to existing plant practices

(4) splitting up requirements currently grouped under a single current specification to more appropriate locations in two or more specifications of the ITS

(5) combining related requirements currently presented in separate specifications of the CTS into a single specification of the ITS.

Table A lists all the administrative changes proposed in the ITS. Table A is organized by the corresponding ITS section discussion of change and provides a summary description of the administrative change that was made, and the CTS and ITS LCO references. The NRC staff reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable, because they are compatible with the Writers Guide and ISTS, do not result in any substantive change in operating requirements, and are consistent with the Commission's regulations.

B. Technical Changes - More Restrictive

The licensee, in electing to implement the specifications of the ISTS, proposed a number of requirements more restrictive than those in the CTS. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or that have additional restrictions that are not in the CTS but are in the ISTS. Examples of more restrictive requirements are placing an LCO on plant equipment which is not required by the CTS to be operable, more restrictive requirements to restore inoperable equipment, and more restrictive SR. Table M lists the more restrictive changes proposed in the ITS. Table M is organized by the corresponding ITS section discussion of changes and provides a summary discussion of the more restrictive changes that were adopted, and the CTS and ITS LCO references. These changes are additional restrictions on plant operation that enhance safety, and are appropriate and acceptable.

An example of a more restrictive technical change is the inclusion of ISTS 3.7.3 on main feedwater isolations valves (MFIVs) in the ITS, where the CTS do not have a separate TS on these valves. The operability of these valves is currently governed by CTS 3.6.3 on containment isolation valves, but ITS 3.7.3 was modified to incorporate the containment isolation requirements on these valves in CTS 3.6.3. A paragraph was added to the Bases of ITS 3.6.3 to state that MFIV operability is covered in ITS 3.7.3.

C. Technical Changes - Less Restrictive (Specific)

Less restrictive requirements include changes, deletions and relaxations to portions of CTS requirements that are not being retained in the ITS. When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on the ISTS. The NRC staff reviewed generic relaxations contained in the ISTS and found them

acceptable because they are consistent with current licensing practices and the Commission's regulations. The PVNGS design was also reviewed to determine if the specific design basis and licensing basis for PVNGS were consistent with the technical basis for the model requirements in the ISTS, and thus provide a basis for the ITS.

A significant number of changes to the CTS involved changes, deletions and relaxations to portions of CTS requirements evaluated as Categories I through IX that follow:

Category I	-	CTS LCO Applicability Changes
Category II	-	CTS Surveillance Frequency Changes
Category III	-	CTS LCO Revised to Address Train Configurations
Category IV	-	CTS Allowed Outage Time Extensions From 24 Hours to 72 Hours
Category V	-	CTS Action requirements for Existing LCO Are Changed
Category VI	-	CTS Surveillance Acceptance Criteria Are Changed
Category VII	-	Other CTS Allowed Outage Time Extensions
Category VIII	-	Elimination of CTS Reporting Requirements
Category IX	-	Relaxation of LCO Requirements

The following discussions address why various technical specifications within each of the above nine categories of information or specific requirements are not required to be included in the ITS.

CTS LCO Applicability Changes *(Category I)*

Reactor operating conditions are used in the CTS to define when the LCO features are required to be operable. CTS applicabilities can be specific defined terms of reactor conditions: hot shutdown, cold shutdown, reactor critical or power operating condition. Applicabilities can also be more general. Depending on the circumstances, the CTS may require that the LCOs be maintained within limits in "all modes" or "any operating mode." However, generalized applicability conditions are not contained in the ISTS, therefore the ITS eliminate CTS requirements such as "all mode" or "any operating mode," replacing them with ITS defined modes or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

In another application of this type of change, the CTS requirements may be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS can be satisfied by exiting the applicability thus taking the plant out of the conditions that require the safety system to be operable. These changes are consistent with the ISTS and changes specified as Category I are acceptable.

CTS Surveillance Frequency Changes *(Category II)*

The CTS and ITS surveillance frequencies specify time interval requirements for performing surveillance requirement testing. Increasing the time interval between surveillance tests in the ITS results in decreased equipment unavailability-due-to-test which increases equipment availability. In general, the ISTS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ISTS is acceptable based on similar design, like-component testing for the system application, and the availability of other TS requirements which provide regular checks to ensure limits are met.

Reduced testing can result in a safety enhancement because the unavailability- due-to-test is reduced; in turn, reliability of the affected structure, system or component should remain constant or increase. Reducing testing is acceptable where operating experience, industry practice, or industry standards such as manufacturers' recommendations have shown that these components usually pass the required surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance frequency changes to incorporate alternate train testing has been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance, e.g., a 31 day air flow test is an indicator that positive pressure in a controlled space will be maintained because this test would use the same fans as the less frequent ITS 36-month pressurization test and industry experience shows that components usually pass the pressurization test. Additionally, surveillance frequency extension can be based on staff-approved topical reports. The NRC staff has accepted topical report changes where topical report analyses bound the plant-specific design and component reliability assumptions. These changes are consistent with the ISTS and changes specified as Category II are acceptable.

CTS LCO Revised to Address Train Configuration *(Category III)*

Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render a safety system incapable of performing its intended design function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of functional capability for a safety system. The intent of ISTS Conditions is to control the inventory of the equipment taken credit for in the safety analysis

such that a sufficient complement of safety systems is available to perform their intended safety function for analyzed accidents. Following this practice allows increased flexibility in plant operations under circumstances when components in different trains are inoperable without compromising safe operation of the plant. However, fundamental requirements for train separation are required to be met.

Specified ISTS Actions and Completion Times for trains are based on availability of redundant operable features, reasonable time for repairs, and low probability of a design basis accident (DBA) occurring during the period features remain inoperable. In general, the ISTS use industry practice or industry standards for restoring trains to operable status. These changes are consistent with the ISTS and changes specified as Category III are acceptable.

CTS Allowed Outage Time Extensions *(Category IV and VII)*

Upon discovery of a failure to meet an LCO, the ISTS specify times for completing required actions of the associated TS conditions. Required actions of the associated conditions are used to establish remedial measures that must be taken within specified completion times (i.e., allowed outage times). These times define limits during which operation in a degraded condition is permitted.

Adopting completion times from the ISTS is acceptable because completion times take into account the operability status of the redundant systems of TS 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. These changes are consistent with the ISTS and allowed outage time extensions specified as Category IV and VII are acceptable.

CTS Action Requirements for ~~Existing LCO~~ Exiting LCOs are Changed *(Category V)*

The CTS require that in the event specified LCOs are not met, penalty factors to reactor operation, such as resetting setpoints, and power reductions shall be initiated as the method to reestablish the appropriate limits. The ITS are constructed to specify actions for conditions of required features made inoperable. Adopting ITS action requirements for ~~existing~~ LCO exiting applicabilities is acceptable because the plant remains within analyzed parameters by performance of required actions, or the actions are constructed to minimize risks associated with continued operation while providing time to repair inoperable features. Such actions add margin to safety or verify equipment status, such as interlock status, for the mode of operation, thereby providing assurance that the plant is configured appropriately or operations that could result in a challenge to safety systems are exited in a time period that is commensurate with the safety importance of the system. Additionally, other changes to TS actions include placing the reactor in a Mode where the specification no longer applies, usually resulting in an extension to the time period for taking the plant into shutdown conditions. These actions are commensurate with industry standards for reductions in power in an orderly fashion without compromising safe

operation of the plant. These changes are consistent with the ISTS and changes specified as Category V are acceptable.

CTS Surveillance Requirement Acceptance Criteria are Changed *(Category VI)*

The CTS require safety systems to be tested and verified operable prior to entering applicable conditions. The ITS provide the additional requirement to verify operability by actual or test conditions. Adopting the ISTS allowance for "actual" conditions is acceptable because TS-required features cannot distinguish between an "actual" signal or a "test" signal. Category VI also includes changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which when combined include operability verification of all TS-required components for the features specified in the CTS. Adopting this format preference in the ISTS is acceptable because SR that remain include testing of all previous features required to be verified operable. These changes are consistent with the ISTS and changes specified as Category VI are acceptable.

Elimination of CTS Reporting Requirements *(Category VIII)*

The CTS include requirements to submit Special Reports when specified limits are not met. Typically, the time period for the report to be issued is within 30 days. However, the ISTS eliminate the administrative control requirements for Special Reports and instead relies on the reporting requirements of 10 CFR 50.73. ITS changes to reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation. Therefore, this change has no impact on the safe operation of the plant. Additionally, deletion of TS reporting requirements not in 10 CFR 50.73 reduces the administrative burden on the plant and allows efforts to be concentrated on restoring TS required limits. These changes are consistent with the ISTS and changes specified as Category VII are acceptable.

Relaxation of LCO Requirements *(Category IX)*

The CTS provide lists of acceptable devices that may be used to satisfy LCO requirements. The ITS reflect the ISTS approach to provide LCO requirements that specify the protective limit that is required to meet safety analysis assumptions for required features. The protective limits replace the lists of specific devices previously found to be acceptable to the NRC staff for meeting the LCOs. The ITS changes provide the same degree of protection required by the safety analysis and provide flexibility for meeting limits without adversely affecting operations since equivalent features are required to be operable. These changes are consistent with the ISTS and changes specified as Category IX are acceptable.

Additionally, in electing to implement the specifications in the ISTS, the licensee also proposed a number of less restrictive changes to the CTS which do not apply to the above Categories of changes, deletions and relaxations of CTS requirements. These changes are discussed below. The associated discussion of change identifier (e.g., L1) is provided for these unique less

restrictive change. These changes are characterized as unique in the L table of PVNGS Less Restrictive Changes.

Section 1.0 - Less Restrictive

L1 The CTS does not include a provision equivalent to the ISTS that combines the definition of channel functional test for both analog and bistable channel, providing requirements for such tests. Combining the definitions for both analog and bistable channels in the ITS allows the bistable channel test signal to be injected "as close to the sensor as practicable" in lieu of "into the sensor" as required by the CTS definition. Injecting a signal at the sensor increases the probability of actuating related circuits that are not being tested in those cases where several logic channels are associated with one sensor. Therefore, performing the test by injecting a signal at the sensor may require (1) jumpering associated logic channels to prevent their initiation during the test or (2) increasing the scope of the test to include the other logic channels. Allowing initiation of the signal close to the sensor provides a complete test of the desired logic channel while reducing the probability of an undesired initiation. This change is a less restrictive requirement for unit operations and is consistent with the ISTS, and is acceptable.

L2 ~~The CTS does not include a definition for core alteration in Section 1.0; therefore, the~~ The CTS does not ~~include the provision in the definition in the ISTS that~~ specifies the specific components (i.e., the movement of fuel, source, or reactivity control components) that comprise a core alteration. Including the ISTS definition in the ITS relaxes the requirement for core alteration ~~from any component to core alterations from the~~ by limiting the definitions of core alterations to the movement of the specific components: fuel, source, or reactivity control components. The ITS definition will not exclude movement of control element assemblies (CEAs) when withdrawn into the upper guide structure, as a core alteration because they are reactivity control components. The ITS definition will exclude the movement of components other than fuel, sources, or reactivity control components as core alterations. The movement or manipulation of other components have a negligible (if any) effect on core reactivity; therefore, there is no need for a restriction on the movement of components other than fuel, sources, or reactivity control components. This change is consistent with the ISTS and is acceptable.

L3 The ITS definition of shutdown margin (SDM) relaxes the requirement in the CTS definition to account for the single control element assembly of highest reactivity worth being fully withdrawn in the determination of the margin. The CTS does not include the provision in the ISTS definition that allows the single control element assembly (CEA) of highest reactivity worth not to be assumed to be fully withdrawn in the calculation of SDM if all the CEAs are verified to be fully inserted by two independent means. This change is acceptable because requiring the CEA of highest reactivity worth to be assumed to be withdrawn when all CEAs have been verified as being fully inserted by

delete

two independent means is overly conservative. This change is consistent with the ISTS and is acceptable.

L4 The ITS definition of channel functional test does not include the following statement in the CTS definition that: "The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy." The intent of the channel functional test is to verify channel operability not to verify setpoints. The ITS deletes the CTS requirement to check setpoints during the test and this change is a less restrictive requirement for unit operations. This change is acceptable because, by definition, setpoints are verified and, if required, adjusted during the performance of a channel calibration. A channel calibration is inclusive of a channel functional test. This change will afford PVNGS the opportunity to remove setpoint verification from the channel functional test and to rely on the channel calibration for this function. The licensee stated that (1) this will be made on an individual basis as analysis shows that setpoint verification performance is not adversely affected when extended out to channel calibration frequencies and (2) the setpoint verification frequency changes are controlled under the 10 CFR 50.59 evaluation process to ensure that any changes receive appropriate review. This change is consistent with the ISTS and is acceptable..

L5 The CTS definition of channel functional test only allows injecting a "simulated signal" into the channel for the test. The CTS does not include a provision in the ISTS definition that allows for a "simulated or actual" signal to be used during channel functional testing. Allowing the use of an "actual" signal is a less restrictive requirement for unit operations; however, some tests are performed by insertion of an actual signal. For others, when a simulated signal is typically used, there is no reason why an actual signal would not suffice for satisfactory performance of the test. Use of an actual signal instead of a simulated signal will not affect the performance of the channel. The ISTS definition will also allow the licensee to take credit for unplanned actuations which used actual signals if sufficient information is collected to satisfy the surveillance test requirements. Operability can be demonstrated in either case since the channel itself cannot discriminate between the "actual" or "simulated" signal. The ITS definition does not change the technical content or validity of the channel functional test. Therefore, this change is consistent with the ISTS and is acceptable.

uneeded extra space

L6 The CTS does not include a provision in the ISTS definitions that allows the channel functional test, the engineered safety features response time, and the reactor protection system response time to be performed by "any series of sequential, overlapping, or total steps so that the entire [channel or response time] is [tested or measured]" to the equipment covered by these definitions. In the CTS, this could be done only for the radiological effluent process monitoring channels for the channel functional test. Allowing the application of this provision in these ISTS definitions to all equipment covered by the channel functional test, the engineered safety features response time, and the reactor protective system response time is a less restrictive change. This

change in these definitions from the CTS to the ITS describes the licensee's application of the definitions at the units, does not change the intent of the channel functional test, the engineered safety features response time, or the reactor protection system response time and ensures that the definitions are applied uniformly to all applicable equipment. The important aspect of performing this test or determining the response times is to test the entire channel or determine the entire response time, and not to specify how the tests are conducted which is what the ISTS definitions allow. This change is consistent with the ISTS and is acceptable.

L7 The CTS definition of Staggered Test Basis is more restrictive than the definition in the ISTS. The CTS definition does not include the ~~provision in the ISTS that eliminates the~~ requirement to test each component at equal intervals. The ~~CST~~ CTS requirement for three pumps to be tested every 92 days on a staggered test basis would require one pump to be tested every 31 days (i.e., equal subintervals within the total interval of 92 days). The ISTS requirement for three pumps to be tested every 31 days on a staggered test basis would require all three pumps to be tested within a 92-day period, with the interval between testing of the components not specified. Removal of the requirement to test components in equal subintervals is a less restrictive change for unit operations. Including this change in the ITS is acceptable because the purpose of staggered testing is to ensure common failures resulting from testing do not render more than one train inoperable and allowing unequal intervals between individual testing of component or systems should be allowed to meet the intent behind staggering the tests. Therefore, this change is consistent with the ISTS and acceptable.

L8 The CTS definition of Operable-Operability is more restrictive than the definition in the ISTS. The CTS definition does not include the provision in the ISTS definition that requires either "normal or emergency" power be available to a system, subsystem, train, component or device for the equipment to be considered operable. The CTS definition only refers to power having to be available to the equipment. Relaxing the requirement for power in the ITS definition of operable to allow a "normal or emergency" source of power constitutes a less restrictive change; however, TSs on ac power sources must be entered which ensures that adequate measures (e.g., cross-train checks to ensure operability of redundant components, systems) are taken so that loss of the safety function of the equipment does not occur and the equipment is capable of meeting its safety analysis requirements even with a loss of a power source. The important aspect of being operable in terms of power is that power is available to the equipment, not what is the power source: normal or emergency. This change is consistent with the ISTS and is acceptable.

Section 3.0 - Less Restrictive

L1 CTS 3.0 is revised to adopt ITS 3.0.6, which provides clarification of required actions for supported systems that are inoperable because of an inoperable support system. CTS 3.0.1 states in part, "... upon failure to meet the Limiting Conditions for Operation, the

associated Action requirements shall be met. 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 except as provided in LCO 3.0.5 and 3.0.6." ITS LCO 3.0.6 clarifies the application of ITS Required Actions for supported systems which are inoperable due to an inoperable support system. ITS LCO 3.0.6 requires the support systems' required Actions entered if their LCOs are not met. Supported systems' LCOs do not have to be entered unless specifically stated in the support systems' Specification. The required Actions entered for the support system ensure that the ability to respond to an accident is restored in an appropriate amount of time. Additional evaluations may be required per the Safety Function Determination Program (SFDP) which assures that no loss of safety function exists, or provides guidance if one does exist. PVNGS current operating practice is to cascade TS, declaring all supported systems inoperable if a supporting system is determined inoperable. This change is acceptable because ITS LCO 3.0.6 contains a provision for performing an evaluation under the SFDP. If a loss of safety function is determined to exist, the applicable LCOs and required Actions are still entered. This change is consistent with the ISTS.

L4 CTS 4.0.3 is revised to adopt remedies in ISTS SR 3.0.3 upon discovery of a missed surveillance. Both CTS 4.0.3 and ITS SR 3.0.3 specify Actions required to be taken upon discovery of a missed surveillance. Upon such a discovery, CTS 4.0.3 requires declaring the LCO not met and entering into the required Actions. If the completion time specified in the Actions is less than 24 hours, the Action requirements may be delayed for 24 hours to permit performance of the surveillance. ISTS SR 3.0.3 allows that upon the discovery of a missed surveillance, the lesser of 24 hours or the specified frequency is allowed to perform the surveillance prior to declaring the LCO not met. Whenever the required Action completion time is greater than 24 hours, CTS 4.0.3 requires immediately declaring the LCO not met and entering the Action (i.e., surveillances must be performed within the allowed Action completion time). ITS SR 3.0.3 allows 24 hours or less to perform the surveillance prior to performing the required Actions. In this case, ISTS SR 3.0.3 is less restrictive than CTS 4.0.3. Allowing time to perform the surveillance prior to entering the required Actions is acceptable since the most probable result of any particular Surveillance being performed is the verification of conformance with the ISTS requirements. The delay period provides time to perform the surveillance before complying with the required Actions. This change is consistent with the ISTS and is acceptable.

L5 CTS 3.0 is revised to adopt ISTS 3.0.5 that allows returning equipment to service under administrative control solely to perform testing to demonstrate its operability or the operability of other equipment. ITS LCO 3.0.5 adds a new LCO Applicability requirement which allows inoperable equipment to be returned to service under administrative controls to perform testing required to demonstrate its operability or the operability of other equipment. This is consistent with current operating practice for performing Surveillance testing, post-modification testing and retest after maintenance to verify operability. ITS LCO 3.0.5 establishes an allowance that is not recognized in

the CTS. It is a less restrictive change because it specifies an exception to ITS LCO 3.0.2. This change is consistent with the ISTS and is acceptable.

Section 3.1.1 - Less Restrictive

L1 CTS SR 4.1.1.1.3 ~~does not contain a provision equivalent to the corresponding ISTS SR which requires the reactivity worth of any CEAs not capable of being inserted be accounted for in the determination of SDM.~~ CTS SR 4.1.1.1.3 requires, if the reactor trip breakers are open and any CEA(s) are fully or partially withdrawn, verifying SDM within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn. ~~ITS 1.1 revises~~ the definition of SDM to require accounting for the reactivity worth of any CEAs not capable of being inserted in the determination of SDM. Therefore, any CEA(s) fully or partially withdrawn, with the reactor trip breakers open, would be accounted for in the ITS SDM Surveillance Requirements. This change is consistent with the ISTS and is acceptable.

Section 3.1.2 - Less Restrictive

L2 CTS SR 4.1.1.2.1a ~~does not contain a provision equivalent to the corresponding ISTS SR which requires the reactivity worth of any CEAs not capable of being inserted be accounted for in the determination of SDM.~~ CTS SR 4.1.1.2.1a requires, if the reactor trip breakers are ~~open~~ and any CEA(s) are ~~fully or partially withdrawn~~, verifying SDM within one hour after detection of the ~~withdrawn~~ CEA(s) and at least once per 12 hours thereafter while the CEA(s) are ~~withdrawn~~. ~~ITS 1.1 revises~~ the definition of SDM to require accounting for the reactivity worth of any CEAs not capable of being inserted in the determination of SDM. Therefore, any CEA(s) ~~fully or partially withdrawn~~, with the reactor trip breakers ~~open~~, would be accounted for in the ITS SDM Surveillance Requirements. This change is consistent with the ISTS and is acceptable.

Section 3.1.5 - Less Restrictive

L1 CTS 3.1.3.1 Actions a, c.2, and c.2 b are revised to adopt ISTS 3.1.5 Actions that do not require verification of SDM. These CTS 3.1.3.1 Actions require that SDM be determined when a CEA is inoperable. The required Actions for ISTS 3.1.5 do not require verification of SDM. During Modes 1 and 2, SDM is assured based on the CEA positions. Under the Combustion Engineering (CE) core design methodology, the changes in fuel and moderator temperature are included in the CEA Power Dependent Insertion Limits (PDIL) which ensures adequate shutdown margin in Modes 1 and 2. When CEAs are inoperable, calculating the SDM is not ~~appropriate~~ required. If the CEAs are above the PDIL, there is adequate SDM and, if CEAs are below the PDIL, restoring the CEAs to within their limits restores SDM. Under the CE design methodology, boration is not ~~an appropriate~~ a required Action for restoring SDM. Therefore, removal of these Actions has no effect on plant safety. This change is consistent with the ISTS and is acceptable.

L3 CTS 4.1.3.1.2 would be changed to delete the requirement to determine operability of each part-length CEA which is inserted in the core by movement of at least 5 inches in any one direction. ISTS 3.1.5 does not include a requirement to exercise part length CEAs because the SDM determination does not take credit for part-length CEAs. Therefore, there is no need to verify part-length CEAs are moveable. This change is consistent with the ISTS and is acceptable.

Section 3.1.7 - Less Restrictive

L1 CTS 3.1.3.6.a.1.b is revised to adopt ISTS 3.1.7 Action C which restricts operation between the long-term, steady-state insertion limits and the transient insertion limits for 14 effective full-power days (EFPDs) per 365 EFPDs ~~intervals~~, instead of per 18 effective power months ~~interval~~. CTS 3.1.3.6.a.1.b restricts operation between the long term steady state insertion limits and the transient insertion limits to (1) 5 EFPDs per 30 EFPDs ~~interval~~ and (2) 14 EFPDs per 18 effective full power months. ISTS 3.1.7 restricts operation between the limits to (1) 5 EFPDs per 30 EFPDs ~~interval~~ and 14 EFPDs per 365 EFPDs ~~interval~~. The interval for the second restriction is being changed from 18 effective full power months to 365 EFPDs. The first requirement restricting operation between the long term steady state insertion limits and the transient insertion limits to 5 effective full power days per 30 effective full power day interval is not changed. The change to the second requirement restricting operation between the long term steady state insertion limits and the transient insertion limits to 14 EFPDs per 365 EFPDs ~~interval~~, is negligible because the effects are very small and are accommodated in the physical uncertainties in the safety analysis, and sufficient time is allowed for power distributions to return to the safety analysis assumptions. Therefore, this change ensures that acceptable limits are placed on operation with flux patterns outside those assumed in the long term core burnup assumptions. This change is consistent with the ISTS and is acceptable.

Section 3.1.10 - Less Restrictive

L1 CTS 3.10.2, on special test exceptions for Modes 1 and 2, is revised to adopt ~~the~~ ISTS 3.1.10. ~~3.1.10 provision that CTS 3.2.1 on linear heat generation rate (LHR) is not required to be met during performance of physics tests in these modes.~~ CTS 3.10.2.b requires maintaining the limits specified in CTS 3.2.1, as determined using CTS 4.10.2.2, during the performance of physics tests, ~~and~~ CTS 4.10.2.2 requires monitoring LHR continuously when power is greater than 20% of rated thermal power (RTP) ensuring the LHR is within the limits of CTS 3.2.1. CTS 3.2.1 requires continuously monitoring LHR above 20% RTP using the core operating limit supervisory system (COLSS) and CTS 4.2.1.2 requires the LHR being verified every 2 hours if COLSS is inoperable. ISTS 3.1.10 does not include a cross reference to maintaining ISTS 3.2.1 requirements on LHR, ~~which are the same requirements in CTS 3.2.1 (i.e., the continuous LHR monitoring required by ISTS 3.2.1 and the 2 hour Frequency allowed by SR 3.2.1.1 when COLSS is inoperable is equivalent to the requirements of CTS 3.2.1).~~ These

requirements are sufficient to identify trends that would result in an approach to the LHR limits. With ITS 3.2.1 applicable above 20% RTP, a cross reference in ITS 3.1.10 to ITS 3.2.1 is not required and does not impact safe operation of the units. This change is consistent with the ISTS and is acceptable.

Section 3.2.1 - Less Restrictive

L1 CTS 3.2.1 Actions a.1 and a.2 are revised to adopt ISTS 3.2.1 Actions A.1 and B.2.1, ~~which require the LHR to be restored to within limits within 1 hour, instead of the 15 minutes in the CTS.~~ CTS 3.2.1, Action a.1, requires that with the LHR not within the LCO limit, "within 15 minutes initiate corrective action to restore the linear heat rate to within the LCO limit within 1 hour." CTS 3.2.1, Action a.2, requires that if an adverse trend in LHR exists, "within 15 minutes initiate corrective action to restore the LHR to within the LCO limit within 1 hour." ISTS 3.2.1, Actions A.1 and B.2.1, require restoring the LHR to within limits within 1 hour. The requirement to initiate corrective action within 1 hour is in ITS 3.2.1. Requiring restoration of the LHR to within limits in 1 hour ensures that prompt action is taken to reduce LHR to below the specified limit. This change is consistent with the ISTS and is acceptable.

Section 3.2.4 - Less Restrictive

L1 CTS 3.2.4 Actions a.1 and a.2.a are revised to adopt ISTS 3.2.4 Actions A.1 and B.1.2, ~~which require the departure from nucleate boiling (DNBR) to be restored to within limits within 1 hour, instead of the 15 minutes in the CTS.~~ CTS 3.2.4 Action a.1 requires that, with DNBR not within the LCO limits, "within 15 minutes initiate corrective action to restore the DNBR to within the LCO limit within 1 hour." CTS 3.2.4, Action a.2.a, requires, that if an adverse trend in DNBR exists, "within 15 minutes initiate corrective action to restore the DNBR to within the LCO limit within 1 hour." ISTS 3.2.4, Actions A.1 and B.2.1, require restoring DNBR to within limits within 1 hour. The requirement to initiate corrective action within 1 hour is in ITS 3.2.4. Requiring restoration of DNBR to within limits in 1 hour ensures that prompt action is taken to reduce DNBR to below the specified limit. This change is consistent with the ISTS and is acceptable.

Section 3.3.2 - Less Restrictive

L6 The ITS clarify the intent of CTS Table 3.3-1 Action 10 requirements to lower the setpoint for the Logarithmic Power Level - High RPS trip in Modes 3, 4, and 5 with the RTCBs closed and the CEDMCS capable of CEA withdrawal. This requirement was added to the CTS by Amendment 98 to the Unit 1 Technical Specifications, Amendment 86 to the Unit 2 Technical Specifications, and Amendment 69 to the Unit 3 Technical Specifications. The requirement was added to ensure assumptions in the PVNGS safety analysis are included in the Technical Specifications. If all four Reactor Coolant Pumps (RCPs) are running the Logarithmic Power Level trip setpoint of 1E -2% RTP will provide the necessary protection. If less than four RCPs are running, the trip setpoint must be

lowered to 1E -4% RTP to provide the necessary protection for a bank CEA withdrawal. The CTS allows an alternate method of providing the necessary trip at 1E -4% RTP by making the Core Protection Calculators (CPCs) operable; however, the CPCs are bypassed at less than 1E -4% RTP. If power increases above 1E -4% RTP the CPCs will automatically be removed from the bypass condition. A trip will then be generated because the CPCs will trip the plant with less than four RCPs running. ITS Table 3.3.2-1, Function 1 removes the option of using the CPCs for protection in MODES 3, 4 and 5. This change is consistent with the PVNGS safety analysis.

Section 3.3.3 - Less Restrictive

L3 CTS Table 3.3-1 Action Statement 6.b.3 requires SRs 4.1.3.1.1, 4.1.3.5, 4.1.3.6, and 4.1.3.7 to be met, except during SR 4.1.3.1.2. The ITS require SR 3.1.5.1 to be performed every four hours. SR 4.1.3.1.1 is the same requirement as ITS 3.1.5.1 (i.e., verify that all CEAs are within 6.6 inches of the other CEAs in the same group). In both the CTS and ITS the frequency of this test will increase, from 12 hours to four hours, while in this action. SR 4.1.3.5 is the verification that all shutdown CEAs are fully withdrawn. SR 4.1.3.6 verifies that all regulating groups are within the Transient Insertion Limit of the Core Operating Limits Report (COLR). SR 4.1.3.7 is the verification that all of the part length CEA groups are within the Transient Insertion Limits of the COLR. In the CTS, SRs 4.1.3.5, 4.1.3.6, and 4.1.3.7 go from a 12-hour interval to a 4-hour interval when the CEA Calculators (CEACs) are inoperable. The ITS does not require a change in frequency for these SRs when the CEACs are inoperable. The 12-hour SR interval is adequate for SRs 4.1.3.5, 4.1.3.6, and 4.1.3.7, because with the control element drive mechanism control system (CEDMCS) in the standby mode as required by this action, the probability of CEA misalignment is reduced. Information independent of the CEAC is available to the operator. If the CEA is fully withdrawn, an upper electrical limit indicating light on the CEDMCS operator module is displayed on the main control board. If the CEA is fully inserted, a CEA bottom light is displayed on the main control board core mimic display. To annunciate improper operating regulating groups, the control board is equipped with a Power Dependent Insertion Limit (PDIL) alarm. SR 3.1.5.2 verifies that two of the three means of indication are available to the operator, for each CEA. SR 3.1.5.1 requires the operator to read the position of each CEA every four hours and compare the position with other CEAs in the group. This would allow the operator to detect any CEA that is not fully withdrawn as required by LCO 3.3.3 ACTION B.2 or if Regulating CEA group #5 is inserted more than 127.5 inches withdrawn. This change is consistent with NUREG-1432.

L4 CTS Table 3.3-1 Action Statement 6.b.2.c states CEA motion is permitted and that CEDMCS must be operated in either the Manual Group or Manual Individual modes. The ITS does not specify the modes that must be used for the CEA motion permitted by the Action statement. This will allow CEA control in the Manual Sequential mode of operation and results in a less restrictive change to CTS limits. CEA control in the Manual Sequential mode of operation is limited to Group #5 to a maximum of 127.5

inches withdrawn for control, according to ITS Action B.2. The sequential modes of operation do not move Group #4 until Group #5 is inserted to 90" withdrawn. Regulating Group #5 cannot be inserted to 90 inches withdrawn under the conditions of this LCO, so Regulating Group #4 will not move in the Manual Sequential mode. The operation of the CEAs in sequential modes will not violate the requirements of ITS required Action B.2, and, therefore, will not result in operation beyond the analytical limits assumed in the Safety Analysis for PVNGS. This change is consistent with NUREG-1432.

delete

Section 3.3.5 - Less Restrictive

L2 CTS Table 3.3-3, Action 14.b includes a requirement that states in part "that the instrument channels and bypasses in Table 3.3-3 shall be OPERABLE." ITS 3.3.5 states that the "instrument and bypass removal channels for each function in Table 3.3.5-1 shall be OPERABLE." The bypass functions are manually enabled when plant conditions allow, and are automatically removed when plant conditions change and the bypass is no longer allowed. The ITS clarify that only the automatic bypass removal function affects the OPERABILITY of the channel. The ability to place a function in the channel bypass manually is needed to prevent unnecessary trips when plant conditions are such that the protective function is not needed. If the channel cannot be placed in a bypass, it does not affect the ability of the channel to provide the trip function when needed; therefore, it does not affect the OPERABILITY of the channel. This change is consistent with NUREG-1432.

Section 3.3.6 - Less Restrictive

L8 CTS SR 4.3.2.1, Table 4.3.2, Table Notations (1) and (3) require a test of the Actuation Logic subgroup relays every 62 days on a staggered test basis. However, relays not tested at power are tested during each Cold Shutdown. ITS SR 3.3.6.2 requires a nine-month frequency of testing to be done on a staggered test basis. This change is justified by Topical Report CEN-403, Revision 1-A "ESFAS Subgroup Relay Test Interval Extension" and its NRC-approved safety evaluation.

Topical Report CEN-403, Revision 1-A and its NRC-approved safety evaluation have been reviewed and are applicable to PVNGS. See Section III.G. The test interval extension is for the Channel Functional Test of the subgroup relays. Instruments at PVNGS are not affected by calibration drift so the plant specific setpoint calculations are not affected by this change. The NRC-approved safety evaluation requires that plants that use Potter and Brumfield MDR relays ensure their commercial grade equipment certification program is adequate to detect failures that are described in the safety evaluation. The licensee has stated that the PVNGS program will detect these failures.

Section 3.3.12 - Less Restrictive

L3 CTS Table 3.3-1 and Table 4.3-1 Item 1.B.2.b requires two channels of Logarithmic Power indication in MODES 3, 4 and 5 when the reactor trip circuit breakers (RTCBs) are open. ITS 3.3.12 requires two boron dilution alarm system (BDAS) alarm channels in these modes to monitor core reactivity and to alert the operator to the start of an inadvertent boron dilution event. Startup channels provide a more accurate indication of core reactivity conditions at the low neutron flux levels in these modes and provide an earlier control room alarm for changes in shutdown neutron flux levels than the Logarithmic power level instrumentation. The ITS does not require Logarithmic Power Level indication in MODES 3, 4 and 5 because the BDAS is a two-channel system designed to monitor shutdown reactivity levels. The BDAS would alert the operator to changes in shutdown reactivity levels long before the logarithmic power channels. The UFSAR and the Design Basis Manual require an alarm to sound 15 minutes before criticality during an inadvertent boron dilution event. Logarithmic channels cannot meet this requirement and, therefore, are not included in this LCO. ITS 3.3.2 requires the OPERABILITY of the logarithmic channels in Modes 3, 4, and 5 when the RTCBs are closed and CEAs are capable of withdrawal.

Section 3.4.10 - Less Restrictive

L1 The CTS is revised to adopt ~~ITS~~ 3.4.10 which requires performance of ~~ITS~~ Action B with two or more pressurizer safety valves inoperable. This prevents unnecessary entry into LCO 3.0.3. Not requiring entry into LCO 3.0.3 is a less restrictive requirement for unit operations and is consistent with the ISTS. This change is acceptable because the actions contained in the pressurizer safety valve ITS adequately address the actions necessary when ~~those~~ valves are inoperable. Therefore, this is an acceptable less restrictive change.

Section 3.5.2 - Less Restrictive

LB1 The ~~reduction~~ in the minimum required nitrogen cover pressure from 254 psig to 260 psig is discussed in Section III.G for ITS 3.5.2.

Section 3.6.3 - Less Restrictive

L8 CTS 4.6.3.3 is being revised to adopt ISTS SR 3.6.3.5, which requires stroke time testing of automatic power-operated containment isolation valves. The ISTS does not require verification of isolation time for non-automatic power-operated valves that are not automatically actuated. The Bases for this SR states that the "isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis." There may be ~~values~~ credited as containment isolation valves that are power operated (i.e., can be remotely operated) and that do not receive a containment isolation signal (e.g., a Part 50 General Design Criteria (GDC) 57 penetration). These power-operated valves do not have an isolation time as assumed in the accident analyses because they require operator action. Therefore, deleting reference to power-

operated valve isolation time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions in the accident analysis. Other power-operated valves and valves required for CSAS that do not receive a CIAS are tested in accordance with the Inservice Testing (IST) Program and are controlled under the 10 CFR 50.59 evaluation process to ensure that any changes receive appropriate review. Therefore, the non-automatic power-operated valves can be deleted from the SR and this change is acceptable.

Section 3.6.5 - Less Restrictive

LB1 The reduction in the maximum containment air temperature from 120 degrees F to 117 degrees F to account for instrument uncertainty is discussed in Section III.G for ITS 3.6.5.

Section 3.6.6 - Less Restrictive

LB1 The change to the applicability of the containment spray system (CSS) specification to specify that reactor coolant system (RCS) pressure must be greater than or equal to 385 psia is discussed in Section III.G for ITS 3.6.6.

LB2 The reduction in the CSS header minimum water level from 115 feet to 113 feet is discussed in Section III.G for ITS 3.6.6.

Section 3.7.12 - Less Restrictive

L2 CTS 3.7.7 Action b, "Control Room Essential Filtration ~~Emergency Air Temperature Control~~ System," for both trains inoperable, is revised to adopt the provision in ISTS 3.7.12 Action D, on the same system and both trains inoperable, which does not require suspending all positive reactivity changes in Modes 5 and 6. Actions for ITS 3.7.12 will suspend core alterations and movement of irradiated fuel; however, positive reactivity changes as a result of changes in boron concentration are allowed within the limits specified in the COLR. In Mode 6, boron concentration is controlled by ITS 3.9.1, which requires suspending positive reactivity additions with boron concentration not within limits. This less restrictive requirement is consistent with the ISTS and is acceptable.

Section 3.8.1 - Less Restrictive

L2 CTS 3.8.1.1 is revised to delete a footnote providing requirements for an inoperable onsite ac electric power system. Footnote 1 to CTS 3.8.1.1 Action b, for one inoperable diesel generator (DG), states that Action b.3 shall be completed if this condition is entered, ~~and~~ Action b.3 requires that there be a determination that the operable DG is not inoperable due to a common-mode failure within 24 hours, or demonstrate the operability of the remaining DG. In essence, the LCO can not be exited in less 24 hours until the common-mode failure evaluation has been completed or until the other DG has been

If the inoperable DG is restored to operable within 24 hours,

run, even though the inoperable DG is made operable. ITS 3.8.1 Action B.3, however, in the event the inoperable DG is restored to operable status prior to completing either the common-mode determination or the operability demonstration, allows the licensee to exit the LCO while the plant corrective action program will continue to evaluate the common-mode failure possibility. ITS 3.8.1 can be exited before completion of the common-mode failure evaluation without having to run the other DG. This is acceptable because the inoperable DG has been made operable and the continued common-mode failure evaluation ~~and demonstration of operability of the other DG is not required~~ will be completed in accordance to the plant corrective action program. This change is consistent with the ISTS and is acceptable.

Section 3.8.5 - Less Restrictive

L2 The CTS is revised to adopt ITS SR 3.8.5.1, which lists which SRs are applicable in order for dc sources to be considered operable. This list includes all of the SRs in ~~IST~~ ITS 3.8.4. The remaining SRs in SR 4.8.2.1 correspond to SR 3.8.6.1 through SR 3.8.6.3. However, ITS SR 3.8.5.1 has a note stating that "The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8." SR 3.8.4.6 verifies that the battery charger supplies specific amps at ≥ 125 volts for ≥ 8 hours. SR 3.8.4.7 verifies that the battery capacity is adequate when subjected to a service test. SR 3.8.4.8 verifies battery capacity through a discharge test. The ITS note prevents the operable dc sources necessary to support the one or more dc electrical power subsystems required by LCO 3.8.10 from being discharged below their capacity to provide the required power supply or otherwise become inoperable during the performance of SRs. These SRs must still be capable of being met (i.e., remain applicable), but actual performance is not required. This change is consistent with the ISTS and is acceptable.

L3 CTS 3.8.2.2 is revised to adopt ITS 3.8.5, "DC Sources - Shutdown," which has an additional required Action A.1 that states, "Declare affected required feature(s) inoperable." This is not in Action a to CTS 3.8.2.2. The addition of this action constitutes a less restrictive requirement for unit operations because it offers an alternative to immediately suspending core alterations, suspending movement of irradiated fuel, and initiating actions to suspend operations involving positive reactivity additions required by Action a to CTS 3.8.2.2. This alternative allows time in which the licensee can assess the impact of the combination of inoperable required features which may in fact not require stopping all work. This alternative ensures that appropriate restrictions are implemented in accordance with declaring the affected required features inoperable. If the licensee determines that this alternative involves unwanted effort, Action A.2 ~~and A.3~~ to ITS 3.8.5 require immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions following the action for CTS 3.8.2.2. The less restrictive change only allows the licensee to determine the effect of the inoperable features and if they warrant immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions. The change is consistent with the ISTS and is acceptable.

Section 3.8.8 - Less Restrictive

L2 CTS 3.8.3.2 is revised to adopt ITS 3.8.8, "Inverters - Shutdown," which has an additional required action A.1 that states, "Declare affected required feature(s) inoperable." This is not in the Action to CTS 3.8.3.2. The addition of this action constitutes a less restrictive requirement because it offers an alternative to immediately suspending core alterations, movement of irradiated fuel, and initiating action to suspend operations involving positive reactivity additions required by the Action to CTS 3.8.3.2. This alternative allows time in which the licensee can assess the impact of the combination of inoperable required features which may in fact not require stopping all work. This alternative ensures that appropriate restrictions are implemented in accordance with declaring the affected required feature(s) ~~LCO required actions~~ inoperable. If the licensee determines that this alternative involves unwanted effort, Action 2 to ITS 3.8.8 requires immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions following the Action for CTS 3.8.3.2. The less restrictive change only allows the licensee to determine the effect of the inoperable features and if they warrant immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions. The change is consistent with the ISTS and is acceptable.

Section 3.8.10 - Less Restrictive

L3 CTS 3.8.3.2 is revised to adopt ITS 3.8.10, "Distribution Systems - Shutdown," which has an additional required Action A.1 that states, "Declare affected required feature(s) inoperable." This is not in the Action to CTS 3.8.3.2. The addition of this action constitutes a less restrictive requirement because it offers an alternative to immediately suspending core alterations, movement of irradiated fuel, and initiating action to suspend operations involving positive reactivity additions required by the Action to CTS 3.8.3.2. This alternative allows time in which the licensee can assess the impact of the combination of inoperable required features which may in fact not require stopping all work. This alternative ensures that appropriate restrictions are implemented in accordance with declaring the affected required feature(s) ~~LCO required actions~~ inoperable. If the licensee determines that this alternative involves unwanted effort, Action 2 to ITS 3.8.10 requires immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions following the Action for CTS 3.8.3.2. The less restrictive change only allows the licensee to determine the effect of the inoperable features and if they warrant immediately suspending core alterations, movement of irradiated fuel, and operations involving positive reactivity additions. The change is consistent with the ISTS and is acceptable.

Section 3.9.2

L1 CTS 3.9.2 Actions a and b are revised to adopt the provision in ISTS 3.9.2 Actions A and B that the actions are required with one or both startup range monitors (SRMs)

being "inoperable," instead of ~~referring to~~ requiring the actions ~~are required~~ with the SRMs being "inoperable or not operating." The Actions for CTS 3.9.2 provide the required actions with one or both SRMs inoperable or not operating, and the Actions for ISTS 3.9.2 provide the required actions with one or both SRMs inoperable. For the modes that the SRMs are required to be operable, the SRMs must be in operation to be considered operable; therefore, it is being redundant to add the phrase "or not operating" to the phrase "inoperable" in these Actions. Not including "or not operating" from the conditions requiring the Actions in ITS 3.9.2 does not alter the application of this specification to the use of SRMs at PVNGS. This change does not impact the safe operation of the plan and is consistent with the ISTS, and is acceptable.

Section 5.0 - Less Restrictive

Plant Superintendent (identified at PVNGS as Department Leader, Operations)

L1 CTS 6.5.2.5 is revised to adopt the provision in ISTS 5.1.1 that the ~~Department Leader~~, or his designee shall review proposed tests and experiments affecting nuclear safety that are not addressed in the UFSAR or the CTS, instead of the Vice President Nuclear Production or his designee in the CTS. CTS 6.5.2.5 requires the Vice President Nuclear Production or his designee for these reviews. ISTS 5.1.1 requires the Department Leader, Operations or his designee for these reviews. This change is consistent with the requirement in CTS 6.5.2.3 that requires the Department Leader, Operations or his designee approve, prior to implementation, modifications to nuclear-safety related structures, systems, and components. Therefore, even though the management level for the review in ITS 5.1.1 is changed to a lower level of management, the requirements for review and approval are the same and the new management level is sufficient for the reviews to be effective in protecting the unit. This change is consistent with the ISTS and is acceptable.

LB1 CTS 6.12.2 is being revised to reduce the distance the dose measurement is made from the source of radioactivity from 18 inches (45 centimeters) to 30 centimeters. CTS 6.12.2 Footnote '**,' states that the dose measurement is made at 18 inches from the source of the radioactivity. The revision to 10 CFR 20 changed the dose measurement from 18 inches to 30 centimeters from the source. Therefore, because ITS 5.7 references the new version of 10 CFR 20, this distance is updated to ~~metric units~~ 30 centimeters. This note is added to ITS 5.7.3, ensuring it is clear where the dose is to be measured with respect to the source. ITS 5.7.3 provides requirements "In addition" to the requirements in ITS 5.7.1. ITS 5.7.1 references the requirements of 10 CFR 20. This change ensures that the ITS 5.7 requirements are consistent with 10 CFR 20 and it is acceptable.

Table L lists all the CTS requirements that have been deleted and which pertain to Category I through IX and to the specific listing of changes discussed above. Table L is organized by ITS section and includes: the section designation followed by the discussion of changes identifier, e.g., 1.1 L.1 (ITS Section 1.1, DOC L.1); a summary description of the changes; the CTS and

ITS LCO references; a reference to the specified change category as discussed above (if applicable); and a characterization of the discussion of changes.

For the reasons presented above, these less restrictive requirements are acceptable because they will not affect the safe operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience, and plant accident and transient analyses, and provide reasonable assurance that the public health and safety will be protected.

D. Relocated Less Restrictive CTS Requirements (Not Entire Specifications)

When requirements within the TS have been shown to give little or no safety benefit, their removal from the TS may be appropriate. This section discusses the relocation of details within the CTS to licensee-controlled documents, instead of the relocation of entire specifications from the CTS to licensee-controlled documents which is discussed below in Section E. In most cases, relaxations previously granted to licensees on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on the ISTS. The NRC staff reviewed generic relaxations contained in the ISTS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The PVNGS design was also reviewed to determine if the specific design basis and licensing basis of PVNGS were consistent with the technical basis for the model requirements in the ISTS, and thus provide a basis for the ITS. A significant number of changes to the CTS involved the removal of specific requirements and detailed information from individual specifications evaluated to be Types 1 through 4 as follows:

Type 1	Details of System Design and System Description Including Design Limits
Type 2	Descriptions of System Operation
Type 3	Procedural Details for TS Requirements and Related Reporting Problems
Type 4	Performance Requirements for Indication-only Instrumentation and Alarms

The following discussions address why each of the following four types of specific requirements and detailed information from individual specifications do not need to be included in the ITS.

Details of System Design and System Description Including Design Limits *(Type 1)*

The design of the facility is required to be described in the UFSAR by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings, and

maintained in accordance with an NRC-approved QA program (i.e., UFSAR Chapter 17 for PVNGS). In 10 CFR 50.59, controls are specified for changing the facility as described in the UFSAR, and in 10 CFR 50.54(a) criteria are specified for changing the QA plan. In the ITS, the Bases also contain descriptions of system design and ITS 5.5.10 specifies controls for changing the Bases. Removing details of system design from the CTS is acceptable because this information will be adequately controlled in the UFSAR, controlled design documents and drawings, or the ITS Bases, as appropriate. Cycle-specific design limits are moved from the CTS to the COLR in accordance with Generic Letter 88-16. ITS Administrative Controls are revised to include the programmatic requirements for the COLR.

Indent →

Descriptions of System Operation *(Type 2)*

Controls specified in 10 CFR 50.54(a) apply to changes to the QA program description.

The plans for the normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures including procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR. In the ITS, the Bases also contain descriptions of system operation. It is acceptable to remove details of system operation from the TS because this type of information will be adequately controlled in the UFSAR, plant operating procedures, or the TS Bases, as appropriate.

QA program description

Procedural Details for Meeting TS Requirements and Related Reporting Problems *(Type 3)*

Details for performing action and surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1, the UFSAR, and the ITS Bases. For example, control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling, and has previously been determined to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this procedural control is consistent with the vast majority of other SRs that do not dictate plant conditions for surveillances. Prescriptive procedural information in an action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is, therefore, required in any event. Other changes to procedural details include those associated with limits retained in the ITS. For example, the ITS requirement may refer to programmatic requirements such as COLR, included in ITS Section 5.5, which specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology.

QA program description

The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled in the UFSAR, plant procedures, ITS Bases, or COLR, as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Similarly, removal of reporting requirements from LCOs is appropriate because ITS 5.6, 10 CFR 50.36, and 10 CFR

50.73 adequately cover the reports deemed to be necessary.

Performance Requirements for Indication-Only Instrumentation and Alarms *(Type 4)*

Indication-only instrumentation, test equipment, and alarms are usually not required to be operable to support TS operability of a system or component unless these items are included in the TS as Accident Monitoring instrumentation. Thus, with the exception of the Accident Monitoring instrumentation, the ISTS do not include operability requirements for indication-only instrumentation and alarms. The availability of such indication-only instruments, monitoring instruments, and alarms, and the necessary compensatory activities if they are not available, are more appropriately specified in plant operational, maintenance, and annunciator- response procedures required by ITS 5.4.1. Removal of requirements for indication-only instrumentation and alarms from the CTS is acceptable because they will be adequately controlled in plant procedures.

Table LA lists the CTS requirements and detailed information removed from individual specifications in the CTS that are relocated to licensee-controlled documents, instead of being transferred to the ITS. Table LA is organized by ITS section and includes: the section designation followed by the discussion of the change identifier, e.g., 2.0 LA1 (ITS Section 2.0, DOC LA1); CTS reference; a summary description of the change; the name of the document that retains the CTS requirements; the method for controlling future changes to the relocated requirements; a characterization of the change; and a reference to the specific change type (i.e., Types 1 to 4), as described above, for not including the information or specific requirements in the ITS. The destination document is included in the table.

The NRC staff has concluded that these types of detailed information and specific requirements are not necessary to be in the ITS to ensure the effectiveness of the ITS to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement: (1) ITS Bases controlled in accordance with 10 CFR 50.59, as stated in ITS 5.5.14 "Technical Specifications Bases Control Program;" (2) UFSAR (which includes the Technical Requirements Manual (TRM) by reference) controlled by 10 CFR 50.59; (3) the Offsite Dose Calculation Manual (ODCM) controlled by 10 CFR 50.59 (4) the Inservice Testing Program controlled ~~in accordance with 10 CFR 50.59, as stated in ITS Section 5;~~ and (5) the QA plan, as approved by the NRC and contained in UFSAR Chapter 17 and other identified sections in the UFSAR, controlled by Appendix B to 10 CFR Part 50 and 10 CFR 50.54(a)(5). For each of these changes, Table LA also lists the licensee-controlled documents, and the TS or regulatory requirements governing changes to these documents.

To the extent that requirements and information have been relocated to licensee-controlled documents, such information and requirements are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, where such information and requirements are contained in LCOs and associated requirements in the CTS, the NRC staff has concluded that they do not fall within any of the

four criteria in the Commission's Final Policy Statement (discussed in Part II of this Safety Evaluation). Accordingly, existing detailed information and specific requirements, such as generally described above, may be deleted from the CTS and not included in the ITS.

The only relocation items in Table LA that are addressed in this Safety Evaluation are those where the destination document is the PVNGS QA program because the program is Chapter 17 of the UFSAR and several other sections in the UFSAR outside of Chapter 17 that are identified in Chapter 17. Normally, the QA program for a plant is a stand-alone document separate from the UFSAR or is located in its entirety within Chapter 17 of the UFSAR.

[Combine first sentence with next paragraph]

RELOCATIONS TO THE QUALITY ASSURANCE PROGRAM

The items in Table LA that are the CTS requirements which the licensee proposed are to be relocated to the PVNGS Quality Assurance Program (QAP) are listed in the table starting on page 32.

By letter dated November 26, 1997, the licensee submitted a request for changes to the CTS which included changes that relocate administrative controls related quality assurance requirements to the QA program in the UFSAR. Specifically, the licensee proposed to relocate the following QA details: (1) review and audit requirements, including requirements for the Independent Safety Engineering Department (ISED); (2) procedural coverage, control of procedures, procedure reviews and procedure approval requirements; and (3) records and record retention requirements to the FSAR QAP description. Relocation of the these QA-related requirements from the CTS to the QAP is consistent with the processes described in Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance."

The regulatory requirements related to the content of the TS are in 10 CFR 50.36 which requires that the TS include administrative controls. Administrative controls "are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure safe operation of the facility in a safe manner." The content of the administrative controls section of TS is information that the Commission deems essential for the safe operation of the facility that is not already adequately covered by other regulations. Accordingly, the staff has determined that requirements that administrative control requirements that are specifically required under 10 CFR 50.36(c)(5) and which are not otherwise necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, can be removed from administrative controls of TS and relocated to licensee-controlled documents, such as the QA program where 10 CFR 50.54(a) provides adequate regulatory control.

By letter dated December 12, 1995, NRC Administrative Letter (AL) 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," the NRC AL 95-06 provided information regarding the relocation of TS administrative controls related to the QA program to assist licensees in considering amendment requests for such relocations.

By letter dated November 26, 1997, the licensee submitted changes to the PVNGS QAP description in the UFSAR that incorporated CTS administrative controls related to QA.

The proposed relocations to the QA program are identified in the table starting on page 32.

contained the final matrix of their proposal

The licensee's letters ~~of November 26 and December 16, 1997, and~~ of February 12, 1998, ~~proposed~~ to relocate QA requirements from the CTS to the QA program. The licensee proposed to relocate the following: (1) QA review and audit requirements, including requirements for the ISED, CTS 6.2.3, 6.5, 6.6.1.b, 6.8.1.g & j, 6.8.4, 6.13.b, 6.14.b, and 6.15.1 to UFSAR Section 13.4; (2) QA procedural coverage, control of procedures, procedure reviews and procedure approval requirements contained in CTS 6.5.2.1, 6.5.2.2, 6.5.2.4, 6.8.1.n, o & p, 6.8.2, 6.8.3, 6.8.4.b, 6.8.4.d, 6.8.4.f, and 6.11 to the UFSAR Section 13.5; and (3) QA records and record retention requirements of CTS 6.10 and 6.14 to UFSAR Section 17.2.6.4. These changes are in accordance with AL 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance." The QA program is the logical program or plan for these relocations because what is being relocated are QA requirements subject to the controls imposed by Appendix B to 10 CFR Part 50 and 10 CFR 50.54(a).

As accepted in the NRC letter of January 30, 1992, the QA program for PVNGS is contained in the UFSAR and not a stand-alone document. Other FSAR sections in addition to FSAR Section 17.2 are part of the QA program. Therefore, to ensure that change control in 10 CFR 50.54(a) for the QA program is maintained, the licensee ~~added~~ a listing to UFSAR Section 17.2 of the other UFSAR sections contain~~ed~~ing ~~in the~~ QA program ~~outside of UFSAR Chapter 17~~ as follows:

will add

outside of UFSAR Chapter 17

commitments

Changes to the quality assurance program description in the FSAR will be made in accordance with 10CFR50.54(a). It should be noted that FSAR section 17.2 does not contain all the quality assurance requirements that are subject to control in accordance with 10CFR50.54(a). Additional sections of the FSAR that contain quality assurance requirements include:

- *Section 1.8 for commitments, alternatives, and exceptions to the quality assurance regulatory guides and standards listed in Appendix 17.2B.*
- *Table 3.2-1 and section 3.6 for information regarding the classification of structures, systems, and components within the scope of the quality assurance program.*
- *Section 13.1 for the organizational structure and specific quality assurance responsibilities of the organizations addressed therein.*
- *Section 13.4 for ~~requirements~~ commitments pertaining to reviews and audits.*
- *Section 13.5 for ~~requirements~~ commitments pertaining to procedural coverage, control of procedures, procedure reviews, and procedural approvals.*
- *Section 18.I.B.1 for Independent Safety Engineering (ISE) ~~requirements~~ commitments provided in response to the recommendations of NUREG-0737.*

The staff has reviewed the changes to the CTS sections listed in the previous table which were relocated to the QA program as described above and determined that the incorporation of the changes into the program are acceptable in that the program continues to satisfy the applicable criteria of Appendix B to 10 CFR Part 50 and ~~in~~ Standard Review Plan Section 17.2 of NUREG 0800. There were no reductions in commitments for the QA provisions which were relocated and any changes noted in the relocation of the above QA provisions were administrative or editorial in nature. Based on the considerations discussed above and the fact that any subsequent changes to the relocated QA provisions from the CTS will be controlled in accordance with 10 CFR 50.54(a), the staff concludes that the relocations to the QA program are acceptable and meet the regulations governing QA programs.

Given that the requirements in the QA program implement the regulations pertaining to the (1) QA review and audit functions, (2) QA procedural coverage, control of procedures, procedure reviews and procedure approval requirements, and (3) QA record and record retention requirements, inclusion of these requirements in the CTS is not necessary to assure safe operation of the facility. These requirements define an administrative framework to confirm that plant activities have been properly conducted in a safe manner. They serve also to define a program that provides senior level utility management with assessments of facility operation and recommends actions to improve nuclear safety and reliability. Based upon the relocation of these provisions to the QA program, it is not necessary to include redundant or additional requirements in the TS administrative controls. A license condition will be issued with the ~~IST~~ ITS conversion to enforce the relocation of these QA requirements from the CTS into the QA program.

The licensee will continue to implement the QAP in accordance with the requirements of 10 CFR Part 50, Appendix B, and commitments to ANSI N18.7 as endorsed by Regulatory Guide 1.33, which provides appropriate controls for the approval of changes to the audit functions and frequencies. Changes to the QA program are controlled in accordance with 10 CFR 50.54(a) which requires prior staff approval before implementation of a program change if the change constitutes a reduction in a program commitment. The staff concludes that this regulatory requirement provides sufficient control for the QA requirements being removed from the CTS.

RELOCATIONS TO THE QA PROGRAM

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
~~6.2.2.e~~ delete →	~~13.1.2.3~~	~~Also corrected TS reference at 9.5.1.5.4.2.~~	~~QAP~~
6.2.3	13.4.4		QAP
6.2.3.1	13.4.4.1		QAP
6.2.3.2	13.4.4.2		QAP
6.2.3.3	13.4.4.3		QAP
6.2.3.4	13.4.4.5		QAP
6.2.3.5	13.4.4.4		QAP
6.4 6.4.1	1.8 - RG 1.8 13.2.2 17.2 App. B	Existing requirements; however not all are QA requirements. UFSAR Sections 1.8 and 17.2 Appendix B are existing compliance matrixes including RG 1.8 for Personnel Selection and Training. UFSAR 13.2.2 in the training section is not part of the QA program as described by the licensee. In addition, Item LA.5 in the LA table also references training and requalification of licensed positions as contained in 10 CFR Part 55.	QAP + OTHER
6.5	13.4		QAP
6.5.1	13.4.2		QAP
6.5.1.1	13.4.2.1		QAP
6.5.1.2	13.4.2.2		QAP
6.5.1.3	13.4.2.3		QAP
6.5.1.4	13.4.2.4		QAP
6.5.1.5	13.4.2.5		QAP
6.5.1.6	13.4.2.6	~~New responsibilities listed in items g and h were added.~~ delete	QAP
6.5.1.7	13.4.2.7		QAP

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
6.5.1.8	13.4.2.8		QAP
6.5.2	13.4.1 13.5.1		QAP
6.5.2.1	13.5.1.1		QAP
6.5.2.2	13.5.1.7		QAP
6.5.2.3	13.4.1.1		QAP
6.5.2.4	13.4.1.2 13.5.1.3		QAP
6.5.2.8	13.4.1.3		QAP
6.5.3	13.4.3		QAP
6.5.3.1	13.4.3.1		QAP
6.5.3.2	13.4.3.2		QAP
6.5.3.3	13.4.3.3		QAP
6.5.3.4 ~~(a-l)~~ (a-i)	13.4.3.4 ~~(a-l)~~ (a-i)		QAP
6.5.3.5	13.4.3.5 13.4.5		QAP
6.5.3.5.a	13.4.5.h		QAP
6.5.3.5.b	13.4.5.i		QAP
6.5.3.5.c	13.4.5.j		QAP
6.5.3.5.d	13.4.5.a		QAP
6.5.3.5.e	13.4.5.p		QAP
6.5.3.5.f	13.4.5.o		QAP
6.5.3.5.g	13.4.5.o		QAP
6.5.3.5.h	13.4.5.k		QAP
6.5.3.5.i	13.4.5.k		QAP
6.5.3.5.j	13.4.5.d		QAP

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
6.5.3.5.k	13.4.5.k		QAP
6.5.3.6	13.4.3.6		QAP
6.5.3.7	13.4.3.7		QAP
6.5.3.8	13.4.3.8		QAP
6.5.3.9	13.4.3.9		QAP
6.5.3.10	13.4.3.10		QAP
6.6.1.b	14.4.2.6.c 13.4.2.8	UFSAR Section 13.4.2.8 is part of the QA plan.	QAP
6.7.1.a	13.4.3.4.e	Only offsite review committee (OSRC) 24 hour notification.	QAP + OTHER
6.7.1.b	13.4.2.6.b/c	Only PRB investigation of the TS violation and review of the licensee event report (LER) required by ITS 2.2.5.	QAP + OTHER
6.7.1.c	13.4.3.4.e/g	Only OSRC review of the TS violation and reportable events requiring 24 hour written notification.	QAP + OTHER
6.8.1.g NOTE (1)	13.4.1.1.1, 13.4.2.6.h	Only PRB review of changes to CPC addressable constants.	QAP
6.8.1.j	13.4.5.k	ITS 5.5.1 and 5.5.4 require programs for the ODCM and radioactive effluent monitoring. ITS 5.4.1.c requires quality assurance procedures for environmental and effluent monitoring. UFSAR Section 13.4.5.k only identifies Regulatory Guides 1.21 and 4.1 as the regulatory guidance for the environmental monitoring quality assurance program. This UFSAR reference will be corrected to cite these RGs for the effluent controls program also.	QAP + OTHER
6.8.1.n	2.5.4.13 Table 2.5-18, 13.5.2.2.K	Existing requirements are contained in UFSAR Section 2.5, procedural coverage requirement ~~are~~ IS added at UFSAR Section 13.5.2.2.K.	QAP in UFSAR Section 13.5.2.2.K

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
6.8.1.o and NOTE	13.5.2.2.L		QAP
6.8.1.p and NOTE	13.5.2.2.M		QAP
6.8.2	13.5.1.2 17.2.6.2.1.2	UFSAR Section 13.5.1.5 applies for the portion of the CTS requiring that programs and procedures be reviewed periodically as set forth in administrative procedures.	QAP
6.8.3	13.5.1.6		QAP
6.8.4	13.4.5	Programs are retained in ITS, described in UFSAR 13.5, or contained in other licensee controlled documents. Audit requirements are relocated to UFSAR 13.4.5 as QA program commitments.	QAP + OTHER
6.8.4.a(audit)	13.4.5.b	Audit requirement only is relocated.	QAP
6.8.4.a(1)	13.4.5.b	Audit requirement only is relocated.	QAP
6.8.4.a(2)	13.4.5.b	Audit requirement only is relocated.	QAP
~~6.8.4.b(program)~~ 6.8.4.b(audit)	~~13.5.2.2.N~~ 13.4.5.f	Audit requirement only is relocated.	QAP
6.8.4.c(audit)	13.4.5.e	Audit requirement only is relocated.	QAP
~~6.8.4.d(program)~~ 6.8.4.d(audit)	~~13.5.2.2.O~~ 13.4.5.b	Audit requirement only is relocated.	QAP
6.8.4.e(audit)	13.4.5.c	Audit requirement only is relocated.	QAP
~~6.8.4.f(program)~~ 6.8.4.f(audit)	~~13.5.2.2.P~~ 13.4.5.a	Audit requirement only is relocated. ~~The relocation wording refers both to parameters and activities.~~	QAP
6.8.4.g(audit)	13.4.5.k	Audit requirement only is relocated.	QAP
6.8.4.h(audit)	13.4.5.k	The audit requirement relocated to QA program section in UFSAR; however, the entire CTS 6.8.4.h section is also being relocated to a licensee-controlled document outside the QA plan.	QAP + OTHER
6.10	1.8 - RG 1.88, 17.2B, 17.2.6.4	UFSAR accepts RG 1.88/ANSI N45.2.9 -1974 as existing requirements.	QAP
6.10.1	1.8 - RG 1.88, 17.2B, 17.2.6.4.A	UFSAR accepts RG 1.88/ANSI N45.2.9 -1974 as existing requirements.	QAP

The program description relocated to UFSAR 13.5.2.2.P will include parameters and activities.

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
6.10.1.a	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.1.b	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.1.c	17.2.6.4.A.1		QAP
6.10.1.d	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.1.e	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.1.f	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.1.g	17.2.6.4.A.2		QAP
6.10.1.h	17.2.6.4.A.3		QAP
6.10.2	1.8 - RG 1.88, 17.2B, 17.2.6.4.A	UFSAR accepts RG 1.88/ANSI N45.2.9 -1974 as existing requirements.	QAP
6.10.2.a	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.b	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.c	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.d	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.e	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.f	17.2.6.4.A.4		QAP
6.10.2.g	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.h	17.2.6.4.A.5		QAP
6.10.2.i	~~17.2.6.4.A~~	~~By reference to ANSI N45.2.9 - 1974, A.6~~	QAP
6.10.2.j	17.2.6.4.A.6		QAP
6.10.2.k	17.2.6.4.A	By reference to ANSI N45.2.9 - 1974, A.6	QAP
6.10.2.l	17.2.6.4.A.7		QAP
6.10.2.m	17.2.6.4.A.8		QAP
6.10.2.n	17.2.6.4.A.9		QAP

1,8 (RG 1.88) 17.2.6.4 17.2 B

By commitment to RG 1.88 and ANSI N45.2.9-1974

Current T.S. 6.0 section	UFSAR section (relocated)	Comments	Program
6.10.2.o	17.2.6.4.A.10		QAP
6.10.2.p	17.2.6.4.A.11		QAP
6.10.2.q	17.2.6.4.A.12		QAP
6.11/6.11.1	12.5.1, 13.5.2.2.A	Existing requirements for procedure changes.	QAP in UFSAR Section 13.5.2.2.A
6.13.b	13.4.2.6.h	PRB review and acceptance of process control program changes. The entire Section 6.13 is being relocated to the QA program.	QAP
6.14.a	17.2.6.4.A.12	Record of reviews of ODCM changes.	QAP
6.14.b	13.4.2.6.g	PRB review of ODCM changes.	QAP
6.15.1	13.4.2.6.h	Only PRB review and acceptance of major changes to the liquid, gaseous, and solid radwaste treatment systems.	QAP + OTHER
Table 3.3-1 Action 2	13.4.2.6.f	Action 2 is (1) place the inoperable channel in bypass or trip within 1 hour, (2) review maintaining the channel in this condition, and (3) the channel shall be returned to operable status no later than the next cold shutdown. UFSAR Section 13.4.2.6.f only covers the second item. Only the PRB review was being relocated.	QAP + OTHER
Table 3.3-3 Action 13	13.4.2.6.f	Action 13 is (1) place the inoperable channel in bypass or trip within 1 hour, (2) review maintaining the channel in this condition, and (3) the channel shall be returned to operable status no later than the next cold shutdown. The second item is removed and technical changes are made to items 1 and 3. UFSAR 13.4.2.6.f covers the second item.	QAP + OTHER
3/4.7.9 Bases	13.4.2.6.h	PRB review and acceptance of snubber accessibility and inaccessibility determinations. Only the PRB review is being relocated.	QAP + OTHER

E. Relocated Entire CTS Specifications

The Commission's Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria in 10 CFR 50.36 may be relocated from an existing TS (an NRC-controlled document) to appropriate licensee-controlled documents. This section discusses the relocation of entire specifications within the CTS to licensee-controlled documents, instead of relocating details from specifications in the CTS to licensee-controlled documents which is discussed in previous Section D. These specification requirements include LCOs, Action Statements (i.e., ACTIONS), and associated SRs. In Appendix A to its application dated October 4, 1996, and its supplements, the licensee proposed relocating such specifications from the CTS to the TRM (which is included in the UFSAR by reference and thus controlled by 10 CFR 50.59). The staff has reviewed the licensee's submittals, and finds that relocation of these requirements to the TRM is acceptable, in that changes to these documents will be adequately controlled by 10 CFR 50.59. These provisions will continue to be implemented by appropriate plant procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

The staff addressed the TS that may be relocated to licensee-controlled documents in Table 2 to Appendix C, Combustion Engineering Technical Specifications, in its letter of May 9, 1988, to J. K. Gasper, Chairman, Combustion Engineering Owners Group (CEOG). The TS proposed by the licensee to be relocated from the CTS to licensee-controlled documents are in agreement with the staff's list in Table 2 of the letter.

The licensee, in electing to implement the specifications of the ISTS, also proposed, in accordance with the criteria in the Final Policy Statement, to entirely remove certain TS from the CTS and place them in licensee-controlled documents noted in Table R of the licensee's submittals. Table R lists all specifications and specific CTS details that are relocated, based on the Final Policy Statement, to licensee-controlled documents. Table R provides the following: the CTS section number, the CTS LCO number, and the CTS LCO title. The NRC staff evaluation of each relocated specification and specific CTS detail presented in Table R is provided below. These evaluations may appear similar to others because, as for example the CTS 3.1.2.1 through 3.1.2.6 below, the specifications being removed from the CTS all involve boration of the RCS.

CTS 3.1.2.1, FLOW PATHS - SHUTDOWN

The requirements of CTS ~~LCO~~ 3.1.2.1 and 4.1.2.1 of Specifications 3.1.2.5a and 3.1.2.5b are being relocated to the TRM. These requirements applied to boration systems and borated water sources for shutdown, respectively. The LCO requires at least one of the boron injection flow paths (i.e., spent fuel pool or refueling water tank) to be operable for Modes 5 and 6. If only the spent fuel pool (SFP) (i.e., Specification 3.1.2.5a) is operable, a flow path is required from the SFP via a gravity feed connection and a charging pump to the RCS. If only the refueling water tank (RWT) (i.e., Specification 3.1.2.5b) is operable, a flow path is required from the RWT via

either charging pump, a high-pressure safety-injection pump, or a low-pressure safety-injection pump. Specifications 3.1.2.5a (SFP) and 3.1.2.5b (RWT) require (1) the SFP and the RWT have a minimum borated water volume of 33,500 gallons; (2) a boron concentration be between 4000 and 4400 ppm boron; (3) a solution temperature is maintained between 60 and 180 °F for the SFP; and between 60 and 120 °F for the RWT.

~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths (including the pathways specified in the CTS specifications being relocated) to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or transient and is not assumed in the accident analysis. In case of a malfunction of the chemical and volume control system (CVCS) (a boration subsystem) which causes a boron dilution event, the automatic response, or the response required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.1 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.1.2.2, FLOW PATHS - OPERATING

The requirements of CTS ~~LCO~~ 3.1.2.2 are being relocated to the TRM. This LCO requires that at least two of the three boron injection flow paths be operable. The flow paths from the RWT or SFP through a charging pump to the RCS are the following: (1) a gravity feed flow path through valve CH-536, (2) a gravity feed flow path through valve CH-327, or (3) a gravity feed flow path through valve CH-164. ~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths (including the pathways specified in the CTS specifications being relocated) to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or other transient and is not assumed in the accident analysis. In case of a malfunction of the CVCS (a boration subsystem) which causes a boron dilution event, the automatic response, or the response required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.2 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.1.2.3, CHARGING PUMPS - SHUTDOWN

The requirements of CTS 3.1.2.3 ~~and~~ 3.1.2.1 are being relocated to the TRM. This CTS LCO 3.1.2.3 requires that at least one charging pump or one high-pressure safety injection pump or

one low-pressure safety injection pump, in a boron injection flow path to the RCS specified in CTS 3.1.2.1 and capable of being powered from an operable emergency power source, must be operable. CTS 3.1.2.1 requires one of the following two flow paths must be operable: (1) if only the SFP is operable (in accordance with CTS 3.1.2.5a), then a flow path is required from the SFP via a gravity feed connection and a charging pump to the RCS and (2) if only the RWT is operable (in accordance with CTS 3.1.2.5b) then a flow path is required from the RWT via either a charging pump, a high-pressure safety- injection pump, or a low-pressure safety-injection pump. CTS 3.1.2.5a (for the SFP) and 3.1.2.5b (for the RWT) requires the following for operability: (1) the SFP and the RWT have a minimum borated water volume of 33,500 gallons; (2) a boron concentration of between 4000 and 4400 ppm boron; and (3) a solution temperature between 60 and 180 °F for the SFP and a solution temperature between 60 and 120 °F for the RWT.

~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths (including the pathways specified in the CTS specifications being relocated) to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or other transient and is not assumed in the accident analysis. In case of a malfunction of the CVCS (a boration subsystem) which causes a boron dilution event, the automatic response, or that required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.3 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.1.2.4, CHARGING PUMPS - OPERATING

The requirements of CTS 3.1.2.4 and 4.1.2.4 are being relocated to the TRM. The specification requires that at least two charging pumps are required to be operable in Modes 1 through 4 for boration and reactivity control. ~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or other transient and is not assumed in the accident analysis. In case of a malfunction of the CVCS (a boration subsystem) which causes a boron dilution event, the automatic response, or the response required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.4 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.1.2.5, BORATED WATER SOURCES - SHUTDOWN

The requirements of CTS 3.1.2.5 are being relocated to the TRM. The specification requires a minimum of one of the following borated water sources (SFP or RWT) to be operable. The LCO requires (1) the SFP ~~and~~ the RWT to have a minimum borated water volume of 33,500 gallons, (2) a boron concentration of between 4000 and 4400 ppm boron; and (3) a solution temperature is maintained between 60 °F and 180 °F be maintained for the SFP ~~and~~ between 60 and 120 °F for the RWT. The requirements on the flow paths are in CTS 3.1.2.1 and 3.1.2.2, and these specifications are also being relocated to the TRM as discussed above.

~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or other transient and is not assumed in the accident analysis. In case of a malfunction of the CVCS (a boration subsystem) which causes a boron dilution event, the automatic response, or the response required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.5 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.1.2.6, BORATED WATER SOURCES - OPERATING

The requirements of CTS 3.1.2.6 are being relocated to the TRM. This includes Figure 3.1-1 on minimum borated water volumes. This specification requires ~~each of the following~~ borated water sources ~~(SFP and RWT)~~ to be operable, ~~the following:~~ (1) the SFP and the RWT to have a minimum borated water volume ~~of 33,500 gallons~~, (2) a boron concentration of between 4000 and 4400 ppm boron; and (3) a solution temperature between 60 and 180 °F be maintained for the SFP and between 60 and 120 °F for the RWT.

~~The~~ Section 3.1 ITS require boration of the RCS when it is needed and boration would involve one or more flow paths to inject borated water into the RCS, with appropriate charging pumps to provide the necessary charging head. The boration subsystem is not assumed to be operable to mitigate the consequences of a DBA or other transient and is not assumed in the accident analysis. In case of a malfunction of the CVCS (a boration subsystem) which causes a boron dilution event, the automatic response, or the response required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before shutdown margin is lost and is not assumed to mitigate this event. Therefore, the requirements in CTS 3.1.2.6 are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

Supplementary protection system pressurizer

PROTECTION

CTS 3.3.1 (TABLE 3.3-1, ITEM I.D), SUPPLEMENTARY ~~PROTECTIVE~~ SYSTEM

Table 3.3-1 lists the reactor protective instrumentation channels and bypasses, and Item 1.D is the supplementary protection system for pressurizer high pressure. The requirements in CTS 3.3.1 on operability, actions, and surveillances for the ~~pressure~~ high pressure channel are being relocated to the TRM. This channel is a supplementary protection channel because it augments the reactor protection against overpressurization by using a separate and diverse trip logic from the reactor protection system for initiation of a reactor trip. This supplementary protection system is not part of the primary success path in the mitigation of a DBA or transient, and this requirement is not essential for responding to a DBA or other transient. Therefore, the requirements Table 3.3.1 Item I.D are not needed in the ITS, and may be relocated to a licensee-controlled document outside of TS, because the Final Policy Statement criteria for including them in the ITS has not been satisfied. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.1 (TABLE 3.3-6, ITEM 1.A), FUEL POOL AREA MONITOR (RU-31)

and alarm/trip at

Table 3.3-6 lists the radiation monitoring instrumentation channels, and Item 1.A is the fuel pool area radiation monitor (RU-31). The requirements in CTS 3.3.6 on operability, actions, and surveillances for this monitor are being relocated to the TRM. This radiation monitor is located in the SFP area and is used to monitor radiation ~~to ensure that the area has not exceeded~~ its specified limits. This monitor is part of the actuation of the fuel building essential ventilation system; however, the radiological dose consequences of a fuel handling accident outside containment have been calculated both with and without the use of the fuel building essential ventilation system, and both are less than one-third of the 10 CFR Part 100 limits. Therefore, this monitor is not used to mitigate a DBA or transient and is not assumed in the safety analysis. The SFP limits do not represent initial condition assumptions of an accident analysis. Although these limits represent operating restrictions and Criterion 2 of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including the requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

could

area radiation monitor alarm/trip

CTS 3.3.3.1 (TABLE 3.3-6, ITEM 1.B), NEW FUEL AREA MONITOR (RU-19)

and alarm at

Table 3.3-6 lists the radiation monitoring instrumentation channels, and Item 1.B is the new fuel area radiation monitor (RU-19). The requirements in CTS 3.3.6 on operability, actions, and surveillances for this monitor are being relocated to the TRM. The new fuel area radiation monitor is located in the new fuel area and is used to monitor radiation in the area ~~to ensure that the area has not exceeded~~ its specified limits. There are no automatic functions that are performed by this monitor, and it is not used to mitigate a DBA or other transient or assumed in a safety analysis. The new fuel area limits do not represent initial condition assumptions of an UFSAR accident analysis. Although these limits represent operating restrictions and Criterion 2

could

radiation monitor alarm

of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including the requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.1 (TABLE 3.3-6, ITEM 1.E), MAIN STEAM AREA MONITORS (RU-139 A&B AND RU-140 A&B)

Table 3.3-6 lists the radiation monitoring instrumentation channels, and Item 1.E is the main steam radiation monitors (RU-139 A&B and RU-140 A&B). The requirements in CTS 3.3.6 on operability, actions, and surveillances for this monitor are being relocated to the TRM. The main steam area radiation monitors are used to indicate when radiation in the area has exceeded its specified limits. There are no automatic functions are performed by these monitors, and they are not used to mitigate a DBA or other transient or assumed in a safety analysis. These area limits do not represent initial condition assumptions of a UFSAR accident analysis. Although these limits represent operating restrictions and Criterion 2 of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including the requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.1 (TABLE 3.3-6, ITEM 3), POST ACCIDENT SAMPLING SYSTEM

Table 3.3-6 lists the radiation monitoring instrumentation channels, and Item 3 is the channels for the post accident sampling system (PASS): The requirements in CTS 3.3.6 on operability, actions, and surveillances for the radiation monitoring instrumentation channels in PASS are being relocated to the TRM. The PASS is used to obtain primary coolant and sump samples following an accident, and could be used to indicate if the RCS is outside its limits; however, there are no DBA or other transient analyses which take credit for the system. Although these limits could represent operating restrictions and Criterion 2 of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including the requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.2, INCORE DETECTORS

The requirements for the incore detectors are being relocated to the TRM. The incore detectors are required to ensure that measurements obtained accurately represent the spatial neutron flux distribution in the core. Although the incore detectors are used by the non-safety-related computerized monitoring system in reactor-related LCOs, they are not relied upon in the

accident analysis, and no DBA or other transient analysis that takes credit for the incore detectors. Although the system could impose operating restrictions because of its use in some LCOs and Criterion 2 of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including the requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.3, (TABLE 3.3-7), SEISMIC MONITORING

(in CTS 3.3.3.3, 4.3.3.3.1 and 4.3.3.3.2)

The requirement for the seismic monitoring instrumentation listed in Table 3.3-7 ~~to be operable and the seismic monitoring instrumentation in the table~~ are being relocated to the TRM. This instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if the plant must be shut down pursuant to Appendix A of 10 CFR Part 100. Since this determination is performed after an event has occurred, it has no bearing on the mitigation of any DBA and is not assumed in the safety analysis. Therefore, this requirement is not essential for responding to a DBA or other transient, and, in accordance with the Commission's Final Policy Statement does not have to be included in the ITS. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.3.3.4, (TABLE 3.3-8), METEOROLOGICAL INSTRUMENTATION

(in CTS 3.3.3.4 and 4.3.3.4)

The requirements on the meteorological monitoring instrumentation channels and the channels listed in Table 3.3-8 are being relocated to the TRM. This instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive effluents from the units to the atmosphere. This instrumentation is not used as an input assumption for any DBA, does not mitigate any event, and is not assumed in the safety analysis. Therefore, the requirements are not essential for responding to a DBA or other transient, and, in accordance with the Commission's Final Policy Statement does not have to be included in the ITS. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

(and 4.3-5)

CTS 3.3.3.6, POST-ACCIDENT MONITORING INSTRUMENTATION

(in CTS 3.3.3.6 and 4.3.3.6)

The requirements for certain post-accident monitoring instrumentation (PAMI) used to follow the course of an accident are being relocated to the TRM. Each individual instrumentation has a specific purpose; however, the general purpose for PAMI is to provide sufficient information to confirm an accident is processing as predicted (e.g., automatic safety systems are performing properly and deviations from the expected accident course are minimal).

The NRC SE on PVNGS conformance to RG 1.97 is documented in a letter dated June 18, 1985, from G.W. Knighton (NRC) to E.E. Van Brunt (ANPP).

all

The application of deterministic selection criteria to post-accident monitoring instrumentation is documented in a letter dated May 9, 1988, from T. E. Murley (NRC) to J. K. Gasper (CEOG). The NRC staff position was that the table of post-accident monitoring instrumentation in the TS should contain, on a plant-specific basis, the RG 1.97 Type A instrumentation specified in the plant's ~~PVNGS~~ SE on RG 1.97, and RG 1.97 Category 1 instruments ~~dated March 5, 1997~~. Accordingly, this position has been applied to the PVNGS RG 1.97 instruments. Those instruments meeting the criteria are retained in ITS. The instruments not meeting the criteria may be relocated from the CTS to licensee-controlled documents.

The CTS RG 1.97 Type A variables are the following: (1) Containment Pressure, (2) Reactor Coolant Outlet Temperature - T hot (Wide Range), (3) Reactor Coolant Inlet Temperature - T cold (Wide Range), (4) Reactor Coolant System Pressure - Wide Range, (5) Pressurizer Water Level, (6) Steam Generator Pressure, (7) Steam Generator Water Level - Wide Range, (8) Reactor Coolant System Subcooling Margin Monitor, and (9) Combustible Gas Control - Hydrogen Monitors.

Additional CTS RG 1.97 Category 1 variables include the following: (1) Neutron Flux Monitor - Power Range, (2) Containment Water Level - Wide Range, (3) Core Exit Thermocouples, (4) Reactor Vessel Water Level, (5) Containment Area Radiation - High Range, (6) Condensate Storage Tank Water Level, (7) Primary Coolant Activity Level, and (8) Containment Isolation Valve Position.

The variables in the CTS that are not RG 1.97 Type A or Category 1 are the Refueling Water Storage Tank Water Level, Pressurizer Safety Valve Position Indication, Containment Water Level - Narrow Range, and Auxiliary Feedwater Flow Rate. It is the requirements for this instrumentation that is being relocated to a licensee-controlled document. This post-accident monitoring instrumentation in the CTS 3.3.3.6, and Tables 3.3-10 and 4.3-7, are not risk-significant because the variables they monitor do not qualify as a Type A or Category 1 variable (i.e., a variable that is important to safety, or needed by the operator so that the operator can perform necessary manual actions). This instrumentation is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Because the selection criteria in the Final Policy statement has not been satisfied for these non-RG 1.97 Type A or Category 1 variable instruments, the LCOs and SRs for these instrumentation may be relocated to a licensee-controlled document outside the ITS. This relocation is acceptable.

CTS 3.3.3.7, (TABLE 3.3-11), LOOSE-PARTS DETECTION INSTRUMENTATION

3.3.3.7

The requirements for the loose-parts detection instrumentation, as specified in CTS 3.3.3.7 and Table 3.3-11, are being relocated to the TRM. This instrumentation ensures the capability to detect loose metallic parts in the reactor primary system, and avoid or mitigate damage to primary system components from the loose parts. The instrumentation is not required for a DBA and is not assumed in the accident analysis. The instrumentation does not indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial

condition of a DBA or transient. Therefore, this instrumentation is not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the instrumentation, the LCOs and SRs for the instrumentation may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.3.3.8, (TABLE 3.3-12), EXPLOSIVE GAS MONITORING INSTRUMENTATION (, 4.3.3.8)

The requirements on the explosive gas monitoring instrumentation in CTS 3.3.3.8 and Table 3.3-12 are being relocated to the TRM. This instrumentation is provided for monitoring and controlling potentially explosive gas mixtures in the Gaseous Radwaste System. The instrumentation provides no automatic action and is not required for any DBA. The instrumentation does not indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, this instrumentation is not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the instrumentation, the LCOs and SRs for the instrumentation may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.4.3.2, AUXILIARY SPRAY (in CTS 3.4.3.2) (4.4.3.2.1, 4.4.3.2.2, and 4.4.3.2.3)

The requirement to have both auxiliary spray valves operable and the SRs to determine operability are being relocated to the TRM. The auxiliary spray valves are used to depressurize the RCS by cooling the pressurizer steam space. The valves can be used as a backup for recovery from a steam generator tube rupture event or from a small break loss-of-coolant accident (LOCA); however, the valves are not the primary success path for mitigating these types of accidents (they) And are not reliable upon all conditions. The auxiliary spray valves are not required for a DBA and are not assumed in the accident analysis. The valves are not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, the valves are not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the auxiliary spray valves, the LCO and SRs for the valves may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.4.6, (TABLE 3.4-2), RCS CHEMISTRY (, 4.4.6)

The requirements on RCS water chemistry in CTS 3.4.6 and Table 3.4-2 are being relocated to the TRM. Poor RCS coolant chemistry contributes to the long-term degradation of system materials of construction and the water chemistry is monitored to reduce the possibility of failures in the RCS pressure boundary caused by corrosion. However, the chemistry monitoring activity is for long-term preventive purposes rather than of immediate importance to the plant operator. Although the limits on water chemistry represent operating restrictions and

Criterion 2 of the Commission's Final Policy Statement criteria for including requirements in the ITS includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.4.8.2, PRESSURIZER HEATUP/COOLDOWN LIMITS

4.4.8.2.1 and 4.4.8.2.2

The requirements in CTS 3.4.8.2 to limit maximum heatup rate (of 200 °F per hour) and maximum cooldown rate (of 200 °F per hour) of the pressurizer, and the associated SRs, are being relocated to the TRM. The pressurizer heatup and cooldown rate limits are to prevent non-ductile failure of the pressurizer and ensure compatibility of operation with the fatigue analysis performed. The limits meet the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and are consistent with the PVNGS structural analysis. ~~LTS are provided for limits on variables consistent with structural analysis results~~; however, these limits do not represent initial condition assumptions of any accident analysis. Although rate limits represent operating restrictions and Criterion 2 of the Final Policy Statement includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.4.9, STRUCTURAL INTEGRITY

The requirement in CTS 3.4.9 for the structural integrity of ASME Code Class 1, 2, and 3 components to be maintained in accordance with SR 4.4.9, and the SR, are being relocated to the TRM. The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the life of the components. These components are monitored so that the possibility of component structural failure does not degrade the safety functions of the systems the components are in. This monitoring activity is of a preventive nature rather than a mitigative nature, and CTS 3.4.9 is directed more toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. This specification is not necessary to ensure immediate operability of safety systems needed to mitigate accidents. It prescribes inspections performed during unit shutdown and is not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for structural integrity, the LCO and SRs for the valves may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.4.10, REACTOR COOLANT SYSTEM VENTS

and 4.4.10

The requirements on the RCS reactor head vent path in CTS 3.4.10 are being relocated to the TRM. The pressurizer vent path is the credited vent path for the steam generator tube rupture

DBA and this vent path satisfies Criterion 3 of 10 CFR 50.36 and, therefore, must be retained in the ITS. The reactor head vent path is manually operated to exhaust non-condensable gases from the reactor vessel head to prevent these gases from inhibiting natural circulation core cooling following any event; however, operation of the reactor head vents is not assumed in any safety analysis. The operation of these vents is included as part of operator actions after an event has occurred, and only if there is indication that natural circulation is not occurring. Because the selection criteria in the Final Policy statement has not been satisfied for the reactor vessel head vent path, the LCO and SRs for the vent path may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.6.4.3, HYDROGEN PURGE CLEANUP SYSTEM

The requirements in CTS 3.6.4.3 and SR 4.6.4.3 on the hydrogen purge cleanup system (HPCS) are being relocated to the TRM. The HPCS is not the primary method of limiting hydrogen with containment following a LOCA. The hydrogen recombiners are the primary method; however, CTS 3.6.4.2 on hydrogen recombiners allows one hydrogen recombiner to be inoperable indefinitely as long as the requirements of CTS 3.6.4.3 are met. With this provision for using the HPCS in CTS 3.6.4.2, the HPCS would be a backup to the hydrogen recombiners. In the ITS 3.6.7 on hydrogen recombiners, however, the ITS require an inoperable hydrogen recombiner(s) to be restored to operable status within 30 days and does not permit the use of the HPCS to allow an inoperable recombiner to remain inoperable. Therefore, in the ITS, the HPCS would no longer be required for containment hydrogen control and the HPCS would not be part of the primary success path for any DBA or other accident. Based on this, the HPCS does not meet the criteria in the Final Policy Statement for inclusion in the ITS, and the requirements in CTS 3.6.4.3 and SR 4.6.4.3 do not have to be included in the ITS and can be relocated to a licensee-controlled document outside the CTS. The TRM is an acceptable licensee-controlled document and the relocation is acceptable.

CTS 3.7.2, STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The requirements in CTS 3.7.2, including SR 4.7.2, are being relocated to the TRM. The limits on the steam generator pressure and temperature ensure that pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits which will prevent steam generator brittle fracture; however, these limits do not represent initial condition assumptions of any steam generator accident analysis. Although these limits represent operating restrictions on the steam generators and Criterion 2 of the Final Policy Statement includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents and transients. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.7.9, SNUBBERS

Snubbers

The requirements in CTS 3.7.9 and SR 4.7.9, including Table 4.7-2 and Figure 4.7-1, for hydraulic and mechanical snubbers are being relocated to the TRM. Snubbers are designed to allow thermal expansion of piping systems during normal operation while limiting movement during seismic events or other transients. A snubber failure is not an initial condition of any DBA or transient, and there are no analyses in the UFSAR which evaluate the consequences of such failures. Snubbers are not part of the primary success path in the mitigation of a DBA or transient. Snubbers are not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, the ~~valves~~ are not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for snubbers, the LCO and SRs for snubbers may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.7.10, SEALED SOURCE CONTAMINATION

The requirements on seal source contamination in CTS 3.7.10, and SRs 4.7.10.1 and 4.7.10.2 are being relocated to the TRM. The limitations on removable contamination from sealed sources are based upon 10 CFR 70.39(c) limits for plutonium and ensure that leakage from these sources will not result in exceeding allowable limits if ingested or inhaled. These limits are not related to safe operation of the units, and are not required to mitigate the consequences of any DBA or other transient. Although these limits represent operating restrictions on the sealed sources and Criterion 2 of the Final Policy Statement includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents and transients. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.8.1.3, CATHODIC PROTECTION

The requirements in CTS 3.8.1.2 and SR 4.8.1.2 for the cathodic protection system are being relocated to the TRM. The cathodic protection system (CPS) is associated with the diesel generator fuel oil storage system that resides underground; however, the system is not taken credit for in any DBA or other transient. The system is not part of the primary success path in the mitigation of a DBA or transient. It is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, the system is not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the CPS, the LCO and SRs for cathodic protection may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.8.4.1, CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

The requirements in CTS 3.8.4.1 and SR 4.8.4.1 for containment penetration conductor overcurrent protection (CPCOP) devices is being relocated to the TRM. The primary and backup CPCOP devices prevent degradation of electrical penetrations and penetration conductors by de-energizing the affected circuit when an overcurrent condition exists and ensure the pressure integrity of the containment penetration through which the circuit passes. Containment penetration degradation should be identified during containment leak rate tests performed in accordance with 10 CFR Part 50, Appendix J. No DBA or other transient takes credit for these devices and the devices are not part of the primary success path in the mitigation of a DBA or transient. The devices are not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, the devices are not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the CPCOP devices, the LCO and SRs for the devices may be relocated to a licensee-controlled document outside the ITS. The testing and maintenance of the devices can be adequately controlled by the TRM, a licensee-controlled document. This relocation is acceptable.

CTS 3.8.4.2, MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

The requirements in CTS 3.8.4.2 and SR 4.8.4.2.1 for motor-operated valves thermal overload protection and bypass devices is being relocated to the TRM. The thermal overload protection and bypass devices ensure that the devices will not prevent safety-related motor-operated valves from performing their safety function during an accident. No DBA or other transient takes credit for these devices and the devices are not part of the primary success path in the mitigation of a DBA or transient. The devices are not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Therefore, the devices are not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for the CPCOP devices, the LCO and SRs for the devices may be relocated to a licensee-controlled document outside the ITS. The testing and maintenance of the devices can be adequately controlled by the TRM, a licensee-controlled document. This relocation is acceptable.

CTS 3.9.3, DECAY TIME

The requirements in CTS 3.9.3 and SR 3.9.3 on the decay time that the reactor must be subcritical before there is movement of irradiated fuel in the reactor core are being relocated to the TRM. This LCO requires 100 hours to elapse to allow the radioactive decay of the short-lived fission products. The screening criteria for including the requirements in the ITS have been satisfied for Criterion 2 since decay time is consistent with the assumptions used in an accident analysis; however, the activities necessary to be performed at PVNGS before commencing movement of irradiated fuel ensure that 100 hours of subcriticality will elapse

before there is movement of irradiated fuel in the core. Therefore, the decay time LCO and SRs may be relocated to a licensee-controlled document outside TS, because the Final Policy Statement criteria have not been satisfied. The TRM is an acceptable licensee-controlled document and the relocation is acceptable.

CTS 3.9.5, COMMUNICATIONS

The requirements in CTS 3.9.5 and SR 4.9.5 to maintain direct communications between the control room and personnel at the refueling station during refueling are being relocated to the TRM. Communications between the control room and personnel performing core alterations are maintained to ensure prompt notification of significant changes in the plant status or core reactivity condition during refueling. Additionally, these communications allow for coordinating activities that require interactions between control room and containment personnel. However, no credit is given for this communication in a DBA or other accident. Therefore, this communication is not essential for responding to a DBA or other transient. Because the selection criteria in the Final Policy statement has not been satisfied for this communication, the LCO and SRs for such communication may be relocated to a licensee-controlled document outside the ITS. The TRM is an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.9.6, REFUELING MACHINE

The requirements in CTS 3.9.6 and SR 4.9.6.1 for the refueling machine, to ensure a minimum capacity of 3590 pounds and an overload cutoff limit of no more than 1600 pounds, are being relocated to the TRM. Administrative controls exist to ensure that the equipment used to handle fuel within the reactor pressure vessel will function as designed and that the equipment has sufficient load capacity for handling fuel assemblies or control rods or both. The refueling machine is designed with interlocks to provide overload limits to prevent damage to refueling equipment and fuel assemblies. These limits are not taken credit for to mitigate the consequences of a DBA; nor do these limits represent initial condition assumptions of an accident analysis. Although these limits represent operating restrictions and Criterion 2 of the Final Policy Statement includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents and transients. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.9.7, CRANE TRAVEL - SPENT FUEL POOL STORAGE BUILDING

The requirements in CTS 3.9.7 and SR 4.9.7 for the SFP crane limits are being relocated to the TRM. CTS 3.9.7 restricts loads in excess of 2000 pounds not to travel over fuel assemblies in the SFP to limit the effect of a dropped load to (1) the gap radioactivity from a single irradiated fuel assembly and (2) no critical array from any possible distortion of the fuel in the SFP racks. The administrative monitoring of loads moving over the fuel storage racks serves as a backup

to the crane interlocks and physical stops in SR 4.9.7. Although CTS 3.9.7 supports the maximum refueling accident assumption in the DBA, the ~~cane~~ crane travel limits are not monitored and controlled during operation; they are checked on a periodic basis to ensure their operability. Although this limit represents operating restrictions and Criterion 2 of the Final Policy Statement includes operating restrictions, Criterion 2 applies only to those operating restrictions required to preclude analyzed accidents and transients. Therefore, the Final Policy Statement criteria for including these requirements in the ITS is not met. The TRM is an acceptable licensee-controlled document for this information. This relocation is acceptable.

CTS 3.9.12, FUEL BUILDING ESSENTIAL VENTILATION

The requirements in CTS 3.9.12 and SR 4.9.12 for the fuel building essential ventilation system (FBEVS) are being relocated to the TRM. The FBEVS reduces the radiological consequences of a fuel handling accident by using high-efficiency particulate air filters and charcoal absorbers ~~before discharging the radioactivity released from the accident to the atmosphere.~~ The radiological consequences of a fuel handling accident outside the containment have been calculated both with and without the use of the fuel building essential ventilation system, and both are less than one-third of the 10 CFR Part 100 limits. The FBEVS is not part of the primary success path in the mitigation of a DBA or other transient. Public health and safety is adequately protected by placing these requirements in the TRM, licensee-controlled documents. The Final Policy Statement criteria for including these requirements in the ITS is not met. The relocation is acceptable.

CTS 3.10.3, SPECIAL TEST EXCEPTION - REACTOR COOLANT LOOPS

The requirements in CTS 3.10.3, SRs 4.10.3.1, 4.10.3.2, and 4.10.3.3 for reactor coolant loops and the requirements are being relocated to the TRM. This special test exception permitted reactor criticality with fewer than four RCPs in operation during performance of startup physics tests provided certain conditions were met, including that at least one RCP in each loop is in operation. The licensee has stated that this special test exemption is no longer used at PVNGS and is not needed to be in the ITS. It is also not taken credit for in any DBA. CTS 3.10.3 does not meet the screening criteria in the Final Policy Statement for inclusion in the ITS; therefore, this test exception LCO and SRs may be relocated outside the CTS to the TRM, an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.10.6, SPECIAL TEST EXCEPTION - SAFETY INJECTION TANKS

The requirements in CTS 3.10.6 and SR ~~4.6.10.1~~ 4.10.6.1 for the safety injection tank (SIT) isolation valves are being relocated to the TRM. This special test exception permits testing the low-pressure check valves if certain conditions are met. The isolation valves are still capable of automatic operation in the event of a SIAS; therefore, system capability has not been affected. The licensee stated that this special test exception is no longer used at PVNGS and is not needed in the ITS. It is not taken credit for in any DBA. CTS 3.10.6 does not meet the screening criteria in the Final Policy Statement for inclusion in the ITS; therefore, this test

exception LCO and SR may be relocated outside the CTS to the TRM, an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.10.7, SPECIAL TEST EXCEPTION - SPENT FUEL POOL LEVEL

The requirements in CTS 3.10.7, and SRs 4.10.7.1 and 4.7.10.2, for SFP level are being relocated to the TRM. The special test exception is for initial fuel load and startup, and permits loading of the core with the SFP pool less than the borated water requirements in CTS 3.1.2.5a and 3.1.2.6a if certain conditions are met. The licensee stated that this special test exception is no longer used at PVNGS and is not needed in the ITS. It is not taken credit for in any DBA. CTS 3.10.7 does not meet the screening criteria in the Final Policy Statement for inclusion in the ITS; therefore, this test exception LCO and SRs may be relocated outside the CTS to the TRM, an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.10.8, SPECIAL TEST EXCEPTION - SAFETY INJECTION TANK PRESSURE

The requirements in CTS 3.10.8, and SRs 4.10.8.1 and 4.10.8.2, for SIT pressure are being relocated to the TRM. The special test exception allows the performance of low-temperature physics tests with the SIT pressure less than the requirements in CTS 3.5.1d. The licensee stated that this special test exemption is no longer used at PVNGS and is not needed in the ITS. It is not credited in any DBA. Because the screening criteria in the Final Policy Statement for inclusion of this special test exception in the ITS have not been satisfied, this test exception LCO and SRs may be relocated outside the CTS to the TRM, an acceptable licensee-controlled document. This relocation is acceptable.

CTS 3.10.9, SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN AND Kn-1 CEDMS TESTING

The requirements in CTS 3.10.9 and SR 4.10.9 for shutdown margin and control element drive mechanisms (CEDMS) testing are being relocated to the TRM. This special test exception allows the performance of such tests preceding startup without the operator having to be concerned as to whether Specification 3.1.1.1 or 3.1.1.2 is applicable as CEAs are moved if specific conditions are met. The licensee stated that this special test exception is no longer used at PVNGS and is not needed in the ITS. It is not taken credit for in any DBAs. Because the screening criteria in the Final Policy Statement for inclusion in the ITS have not been satisfied, this test exception LCO and SR may be relocated to outside the CTS to the TRM, an acceptable licensee-controlled document. This relocation is acceptable.

The relocated CTS requirements discussed above are not required to be in the TS under 10 CFR 50.36 and do not meet any of the four criteria in the Commission's Final Policy Statement. They are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff finds that sufficient controls exist under the regulations sited above to maintain the effects of the provisions in these specifications. The NRC staff has concluded that appropriate controls have

been established for all of the current specifications, information, and requirements that are being moved to licensee-controlled documents.

There is a license condition to make enforceable the transfer of requirements in the CTS into licensee-controlled documents (i.e., documents, such as the Final Safety Analysis Report (FSAR), for which changes to the documents by licensees are controlled by the regulations, such as 10 CFR 50.59).

Following approval and implementation of the ITS by the licensee, the NRC will audit the removed provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient controls exist under the regulations, in particular 10 CFR 50.59. Accordingly, these specifications, information, and requirements, as described in detail in this Safety Evaluation, may be relocated from the CTS and placed in the UFSAR or other licensee-controlled documents as specified in the licensee's application of October 4, 1996, and its supplements which are listed in Section 1.0 of this Safety Evaluation.

F. Control of Specifications, Requirements, and Information Removed from CTS

In the ITS conversion, the licensee will be relocating specifications, requirements, and detailed information from the CTS to licensee-controlled documents outside the ITS. This is discussed in detail in Sections D and E above. The facility and procedures described in the UFSAR and TRM, incorporated into the UFSAR by reference, can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures record are maintained and establishes appropriate control over requirements removed from the CTS and over future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with other applicable regulatory requirements: for example, the ODCM is required by the ITS to be changed in accordance with 10 CFR 50.59; the emergency plan is required to be changed in accordance with 10 CFR 50.54(q); and the QA program is required to be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. Temporary procedure changes are also controlled by 10 CFR 50.54(a). The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the licensee's QA program for PVNGS and such applicable regulations as 10 CFR 50.59.

The licensee submitted the matrix of CTS requirements that are being relocated from the CTS to licensee-controlled documents in its letter dated ~~May 30 and December 16, 1997~~ February 12, 1998, with revisions in letter dated March XX, 1998. The details of the relocated requirements are shown in the LA table for the PVNGS IST conversion which were also submitted in the ~~May 30 and December 16, 1997~~ letters. The licensee stated that, once the PVNGS IST has been implemented, changes to the documents in the matrix to relocate the CTS requirements will processed in accordance with the controls identified in the matrix. The document changes are to have the CTS requirements designated for placement in the UFSAR or the TRM are appropriately reflected in these documents, or that they will be included in the next required update of these documents. This relocation is subject to a license condition established herewith to make the relocation enforceable. The licensee is required to

Insert for page 54

A license condition is imposed with this amendment with the following requirement:

"Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the Table of PVNGS Relocated Details (LA) and table XX [the table of quality assurance-related relocations beginning on page 32 of the draft SE] in the Safety Evaluation enclosed with this amendment, as evaluated in the SE.

maintain an auditable record of, and an implementation schedule for, the procedure changes associated with the ~~development~~ of the ITS. The licensee will maintain the documentation of these changes in accordance with the record retention requirements in the QA program and the TRM.

There is a license condition to make enforceable the transfer of requirements in the CTS into licensee-controlled documents (i.e., documents, such as the Final Safety Analysis Report (FSAR), for which changes to the documents by licensees are controlled by the regulations, such as 10 CFR 50.59).

G. Evaluation of Other Changes Included In The Application For Conversion to Improved Technical Specifications

ITS 3.3.6 CTS SR 4.3.2.1, frequency testing of the engineered safety features actuation system (ESFAS) subgroup relays, was extended in accordance with CE Topical Report CEN-403, Revision 1-A and the associated safety evaluation issued by the NRC.

The proposed TS change would extend the surveillance interval of Surveillance Requirement (SR) 4.3.2.1 for the engineered safety features actuation system (ESFAS) instrumentation subgroup relays from 62 days to 9 months on a staggered test basis, and would change the related Bases.

In Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS [Anticipated Transients Without Scram] Events," dated July 8, 1983, NRC requested that licensees review the reactor protection system (RPS) test intervals to determine if they were consistent with achieving high RPS availability. Subsequently, in Topical Reports CEN-327 and CEN-327, Supplement 1, the Combustion Engineering Owners Group (CEOG) presented an analysis of the effects on core-damage frequency in extending the interval for RPS/ESFAS channel functional test from monthly to quarterly using fault tree analysis. The ESFAS subgroup relays were specifically excluded from the CEN-327 and CEN-327, Supplement 1, evaluations.

To address the ESFAS subgroup relays, in July 1991, the CEOG submitted Topical Report CEN-403, "ESFAS Subgroup Relay Test Interval Extension." In response to staff comments and to several incidents involving Potter & Brumfield (P&B) motor-driven rotary (MDR) dc and ac relays, the CEOG revised the topical report and resubmitted it in September 1995 as CEN-403, Revision 1. The revised topical report addressed extending the ESFAS subgroup relay surveillance test interval to each refueling outage on a staggered test basis using mean time between failure (MTBF) data of ESFAS subgroup relay components as the basis for the change. In a letter dated February 27, 1996, the NRC approved CEN-403, Revision 1-A.

In the safety evaluation approving CEN-403, Revision 1, the NRC staff indicated that licensees requesting the ESFAS subgroup relay surveillance interval extension are to confirm the applicability of CEN-403, Revision 1-A, (the A meaning that NRC has approved the revision) to their specific plants. As shown in CEN-403, Revision 1-A, the MTBF of ESFAS subgroup relays at the PVNGS units is 36 months. Using the licensee's requested 9-month staggered test basis in the PVNGS units, the two trains of subgroup actuation relays will be tested at least every 22.5 months (18 months + the 25 percent TS allowance). This interval is less than the 36-month MTBF reported in CEN-403 Revision 1-A for the PVNGS ESFAS subgroup relays, and is, therefore, acceptable.

The NRC staff also stated in the safety evaluation approving CEN-403, Revision 1, that licensees referencing the safety evaluation are to confirm that the applicable setpoint calculations account for any increase in instrument drift caused by the extended surveillance test interval. The licensee stated that there are no instruments affected by subgroup relay calibration drift; therefore, the plant-specific setpoint calculations are not affected by this change. This justification is acceptable.

In the safety evaluation approving CEN-403, Revision 1, the NRC also imposed additional conditions on plants using P&B MDR relays in ESFAS subgroup relay actuations, as follows:

1. Ensure that the commercial-grade equipment certification program is adequate for detecting the types of failures that are discussed in References 8, 9, 11, and 12 of the safety evaluation report approving CEN-403, Revision 1.

   In response to this condition, the licensee stated that the PVNGS commercial-grade equipment certification program will detect the types of failures identified in the cited references. As discussed in CEN-403, Revision 1, the licensee's experience with P&B relay failures and commercial-grade equipment certification of relays supports this statement, and is, therefore, acceptable. This condition has been met.

2. Ensure that all pre-1990 P&B MDR dc relays and all pre-1992 P&B MDR ac relays have been removed from ESFAS applications. This condition arose as a result of numerous problems with older P&B MDR relay designs in CE-designed plants.

   In response to this condition, as stated in CEN-403, Revision 1, the licensee contracted with two laboratories in 1988 to determine the cause of the P&B MDR relay failures. The problems were resolved by the licensee and P&B with the development of a new style of relay, which the licensee installed in the PVNGS units in 1989. Failure data from the new relay design support the licensee's claim of improved reliability. To preclude installation of the older design P&B relays, the vendor assigned new part numbers to the new design P&B relays used at PVNGS. The plant drawings and bill of materials also have been changed to reflect the new part numbers. The licensee stated that these changes will prevent installation of any pre-1990 (i.e., old design) P&B MDR dc relays (PVNGS does not use P&B MDR ac relays in ESFAS applications). The staff concludes that the licensee

actions to ensure that all pre-1990 P&B MDR dc relays and all pre-1992 P&B MDR ac relays have been removed from PVNGS ESFAS applications are acceptable. This condition has been met.

On this basis, the staff finds that the proposed TS changes for extending the ESFAS subgroup relay functional test interval from 62 days to 9 months on a staggered test interval are in conformance with the NRC-approved Topical Report CEN-403, Revision 1-A, and are, therefore, acceptable.

ITS 3.5.2 CTS LCO 3.5.1, safety injection tank minimum nitrogen cover pressure, was increased to include instrument uncertainties.

The proposed TS change would increase the minimum required nitrogen cover pressure for the safety injection tanks (SITs) from 254 psig to 260 psig, and would change the related Bases. The change resulted from new instrument uncertainty values associated with the pressure transmitters in the SIT pressure indication loops. The 235-psig minimum SIT cover pressure used in the analysis (i.e., the analytical limit) was not changed. The licensee's reanalysis noted that the uncertainty associated with the instruments used to measure the minimum nitrogen cover pressure could be as high as 22.6 psig. Previously, only 19 psig was allocated for the instrument uncertainty. Therefore, the minimum nitrogen cover pressure needs to be revised to specify 260 psig to ensure that the analytical limit was not compromised. This change is only applicable to Modes 3 and 4 when the pressurizer pressure is less than 1837 psia.

Each unit has four SITs which supply water to the reactor vessel during the blowdown phase of a large-break LOCA, provide inventory to help accomplish the refill phase that follows thereafter, and provide RCS makeup for a small-break LOCA. The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas and are passive components because no operator or control action is required for them to perform their function. The internal tank pressure is sufficient to discharge the tank contents to the RCS, if RCS pressure decreases below the SIT pressure.

In Modes 3 and 4, with pressurizer pressure less than 1837 psia, CTS 3/4.5.1 requires either (a) four SITs, each with minimum and maximum borated water volumes of 962 cubic feet and 1914 cubic feet, respectively or (b) three SITs, each with minimum and maximum borated water volumes of 1415 cubic feet and 1914 cubic feet, respectively. The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow one less than the number of required-operable SITs to partially recover the core before significant cladding melting or zirconium-water reaction can occur following a LOCA. The need to ensure that one less than the required SITs is adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost through the break during the blowdown phase of a LOCA. This is to meet single failure criteria.

A minimum nitrogen cover pressure requirement (the subject of this evaluation) ensures that the SIT gas volume will generate discharge flow rates during injection that are consistent with

those assumed in the safety analyses. A minimum pressure of 235 psig was used in the PVNGS analyses.

The analytical limit for the minimum pressure of the nitrogen cover is 235 psig. This limit was not changed in the licensee's reanalysis and will not be changed by this evaluation. However, the licensee's latest calculation for the indicator loop uncertainties showed that for normal conditions, a total of the instrument loop uncertainties for the control room indication of the nitrogen pressure will be +22.6 psig and -25.2 psig. Since the new uncertainty values are larger than the present ones, the licensee proposed to change the limiting value of the control room indicated pressure from 254 psig to 260 psig to assure that the actual pressure in the tank will always be ≥ 235 psig, the analytical limit value. If the value of the indicated pressure in the control room is held at 260 psig, the actual pressure in the tank will be between 237.4 and 285.2 psig. The lowest value of the actual pressure in the tank is more conservative than the analytical limit of 235 psig.

On this basis, the staff finds it acceptable to increase the minimum required nitrogen cover pressure for the SITs from 254 psig to 260 psig.

ITS 3.6.5 CTS LCO 3.6.1.5, containment air temperature, reduced to incorporate instrument uncertainties.

The proposed TS change would revise the maximum containment air temperature from ≤ 120 °F to ≤ 117 °F, and would change the related Bases. The licensee's reanalysis noted that as much as 3 °F of uncertainty may exist for the instruments that monitor containment air temperature. The 120 °F is the analytical value utilized as an upper bound initial condition in the containment safety analyses and that 3 °F has been determined to be an appropriate plant-specific value to account for instrument uncertainty. The change to the CTS is needed to ensure that the 120 °F upper limit is not exceeded. This change is a result of the licensee's revised analysis which incorporated instrument uncertainties in the analysis.

The staff has reviewed the licensee's proposed change and finds that the change is more restrictive and will serve to ensure that the maximum normal containment operating temperature does not exceed the upper-limit value assumed in the containment peak accident pressure and temperature safety analysis. The containment analyses assume that containment passive heat sinks are initially at a temperature of 120 °F (Reference: UFSAR Section 6.2.1.1.3). If no allowance is provided for instrument uncertainty, there is a potential to operate with a containment temperature of 123 °F, in which case the containment heat sinks would not suppress the containment pressure to the extent credited in the safety analysis.

On this basis, the staff concludes that the design basis accident is properly bounded by the analyses as described in the licensee's UFSAR and is, therefore, acceptable.

ITS 3.6.6 CTS LCO 3.6.2.1, containment spray system applicability, revised to specify "in Modes 1, 2, 3, and 4*" with the asterisk meaning ~~"only when shutdown cooling is not in operation."~~ "when RCS pressure is ≥ 385 psia."

The proposed TS change would revise the Mode 4 operability requirements for the CSS, and would change the related Bases. ~~The requirements would be revised to eliminate the need for operators to enter an emergency shutdown action requirement during a routine shutdown when the CSS is intentionally made inoperable.~~ Replace with Insert A

The CSS at PVNGS consists of two independent trains. Each train comprises a containment sump, a spray pump, ~~a chemical addition pump,~~ a shutdown cooling/spray heat exchanger, a spray header with distribution piping and nozzles, and associated valves and instrumentation. The two trains share the refueling water storage tank (RWST) as the initial source of spray fluid. ~~and a common chemical (hydrazine) addition tank.~~ Each train is capable of automatically performing the post-accident containment cooling function when actuated by a containment spray actuation signal (containment high-high pressure). The CSS post-accident safety function is to prevent the containment design pressure from being exceeded in a design-basis loss of coolant accident (DBA-LOCA), and to reduce the containment pressure to less than one-half the peak accident pressure within 24 hours. (The containment fan coolers are not relied upon for post-accident containment cooling.) Some CSS components are also used in the shutdown cooling system (SDCS).

In addition to the containment cooling safety function, the spray system provides a hydrazine injection to control pH for iodine removal safety function. It can also be aligned to augment the shutdown cooling system when the reactor coolant system temperature exceeds 200 °F.

The CSS is a containment support system and the only post-accident (i.e., safety-grade) containment cooling system. As a containment support system, its operability requirements are established on the basis of consistency with containment operability requirements. The PVNGS primary containment is currently required to be operable whenever the reactor is in Mode 4 (350 °F less Tavg less 200 °F) with the vessel head tensioned and residual heat in the core is not being removed by the shutdown cooling system (SDCS). The SDCS may be used when the RCS pressure exceeds 450 psi and the RCS temperature is less than 350 °F. In Mode 4, a LOCA is a postulated event and the containment system should be fully operable. In Mode 5 (RCS temperature ≤ 200 °F) containment operability is not required and the CSS may, therefore, intentionally be made inoperable for maintenance or to reduce the potential for an inadvertent spray activation.

Mode 5 operation requires use of the SDCS. Because of the low design pressure, the SDCS can only be used for RCS cooling if the RCS pressure is less than 385 psia. In Mode 2, the RCS is fully pressurized and isolated from the SDCS. Because of these operational limitations, there are periods of time during which the SDCS is operated in Mode 4 in order to effect a plant transition between Modes 3 and 5. At PVNGS, the CSS/SDCS heat exchangers are shared with the CSS because the CSS heaters are branch lines from the heat exchanger SDCS outlet lines. In order for the SDCS and the CSS to be operable concurrently, the SDCS flow must

Replace with Insert B

Inserts for page 59

Insert A

The requirements would be revised to change the Mode 4 applicability from the CTS requirement of "only when shutdown cooling is not in operation" to "when RCS pressure is $\geq$ 385 psia." This provision will enable the CSS to be realigned to the shutdown cooling system during a plant shutdown without requiring an unnecessary entry into an Action Condition during the realignment.

Insert B

CTS LCO 3.6.2.1 requires two containment spray systems (CSS) to be operable in Modes 1, 2, 3, and in Mode 4 only when shutdown cooling is not in operation. This change will permit the CSS to be inoperable during Mode 4 when RCS pressure is below 385 psia and the SCS is not in operation. This provision will allow the CSS to be realigned to the SCS during a shutdown after RCS pressure is reduced to below 385 psia.

As described in the Bases for CTS 3/4.6.2.1, the operability of the containment spray system ensures that the containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses. The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned to other inoperable ESF equipment.

In Mode 4, when the RCS temperature is less that 350° F and pressure less than 385 psia, the shutdown cooling system could be placed in operation and the CSS could be inoperable in accordance with CTS 3.6.2.1 Applicability. This change will allow the CSS to be inoperable when the RCS temperature is less that 350° F and pressure less than 385 psia, even if the shutdown cooling system is not in operation. The energy available to initiate a LOCA and be released to the containment atmosphere in a DBA is the same whether or not shutdown cooling is in operation. Therefore, this change to the CTS 3.6.2.1 applicability would not involve any change to the probability or consequences of a LOCA. This change would allow the CSS to be removed from service and aligned to the shutdown cooling system during shutdown and depressurization without requiring an unneeded entry into an Action Condition.

bypass the SDCS heat exchangers. This loss of decay heat removal capability is undesirable. The arrangement thus dictates that, during a normal shutdown, the CSS be taken out of service before placing the RCS on shutdown cooling. During the interim period, the CSS must be declared inoperable, and the plant must implement administrative actions and reporting requirements associated with an emergency shutdown due to a total loss of the containment cooling post-accident safety function.

The proposed change to the CTS would essentially give performance to the decay heat removal function of the SDCS heat exchangers over the containment cooling function during mode transitions, and eliminate the requirement to take the emergency administrative actions. The licensee has determined that, due to the reduced fission product inventory in the core during such periods, the temporary loss of spray system operability during Mode 4 does not constitute a safety question or concern.

On this basis, the staff finds the proposed changes to the CSS operability requirements acceptable.

ITS 3.6.6 CTS SR 4.6.2.1.c, containment spray header piping water level, reduced to include instrument uncertainty.

The proposed TS change would reduce the minimum water level of the CSS header from 115 feet to 113 feet to include instrument uncertainty, and would change the related Bases. The CSS is described in the previous discussion on revising the Mode 4 operability requirements for the system. The CSS experiences evaporation and leakage which causes operations personnel to enter the containment periodically to refill the CSS header with water to ensure that the header remains filled with water above the 115-foot level. Having the required water level 2 feet lower will reduce the number of times that operations personnel have to enter the containment to perform this maintenance.

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UFSAR section 6.5.2.8, paragraph (RA) 7.14

The primary containment has been analyzed for the post-accident pressure and temperature response to a LOCA. Such safety analyses are performed to assure that a worst-case LOCA would not overstress the containment. One of the assumptions made in performing such analyses is that at least one CSS train will be initiated, and full flow established, within a minimum flow initiation time delay (i.e., 32 seconds for PVNGS per ~~Combustion Engineering Standard Safety Analysis Report (CESSAR) System 80, Appendix 6A, paragraph 6.3~~). One of the factors that affects the flow initiation time is the system fill time (i.e., the delay that occurs because of the time it takes to fill the normally empty portion of the spray distribution headers and the distribution piping containing the spray nozzles). A reduction in the level or amount of spray solution in the spray headers (riser) will increase the spray system response time, thereby delaying containment depressurization.

CTS 4.6.2.1.c currently requires that the spray headers be maintained full to at least the 115-foot level. Level indicators and alarms are provided, and surveillance tests are periodically performed to assure that the water level has not dropped. The level could drop from leakage or

evaporation. The licensee requested to lower the minimum allowable water level from 115 feet to 113 feet in the headers. This will establish a wider operating band (2 additional feet) and allow longer periods between the addition of makeup water to replace water lost by evaporation or leakage back to the same maximum level. This will reduce the number of times operating personnel have to enter the containment to perform fluid makeup.

The staff calculated that the increase in the spray system initiation delay due to having to fill 2 additional feet of riser pipe (because of the reduced minimum level from 115 feet to 113 feet) would be approximately 1/8 second. This increase in the delay is considered analytically insignificant. Therefore, the staff concludes that the increased delay in CSS operation in an accident due to the reduced level of fluid in the headers would not interfere with the capability of the CSS to cool the containment under post-accident conditions. The staff finds that the change to the minimum spray header water level from 115 feet to 113 feet is acceptable.

ITS 5.0 CTS SR 4.6.4.3.d.1, allowable pressure drop across the hydrogen purge filtration unit, was revised as a result of a revised analysis.

The proposed TS change would reduce the allowable pressure drop across the hydrogen purge exhaust air filtration unit for the hydrogen purge cleanup system from 8.4 inches of water gauge to 2.26 inches of water gauge and would change the related Bases. The change results from the licensee's revised analysis, which identified that the flowrate was previously changed from 1000 cfm to 50 cfm and that the pressure drop was not revised to correlate to the lower flowrate. This change is needed to specify the correct pressure drop in the TS. The allowable pressure drop of 2.26 inches of water gauge and the corresponding flowrate change from 1000 cfm to 50 are necessary in order to correlate these parameters to the design-basis input assumptions in the post- LOCA hydrogen generation analysis.

The current TS 4.6.4.3.d.1 lists the maximum pressure drop allowed for the hydrogen purge exhaust air filtration Unit for the hydrogen purge cleanup system (HPCS) as 8.4 inches of water gauge. This system performs a cleanup role as the title of the system implies. However, the main system that satisfies the requirements of hydrogen control for DBAs are two hydrogen recombiners that are shared among the three units.

As specified in 10 CFR 50.44, the HPCS is not required for design-basis hydrogen control. However, it does serve as a non-safety backup system if beyond-DBA conditions were to occur. In the unlikely event that both recombiners fail, the hydrogen purge subsystem would be manually put into operation when hydrogen concentration reaches 3.5 volume percent.

As part of the design basis verification project, the licensee determined that the maximum pressure drop for the HPCS is 2.26 inches water gauge as opposed to the current specification of 8.4 inches. The difference seemed to have occurred during the original licensing process. System flowrate was revised and lowered from 1000 scfm to 50 scfm; however, the pressure drop was not lowered. The net result is to have a maximum pressure drop that is too high by almost a factor of 4. Maintaining this large value would conclude operability with excess

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pressure drop across the filter. The change requested would restore the testing criteria to a meaningful test and ensure that the design-basis requirements are maintained for the HPCS.

Until the TSs are approved, the licensee has installed administrative controls to ensure that the testing requirements use the lower maximum pressure drop value of 2.26.

On this basis, the staff concludes that during the time of licensing, the system requirements were modified without changing the allowable pressure drop. As a result, the requested changes to the allowable pressure drop across the filter of the HPCS simply correct a long-term error. The staff has reviewed the requested TS changes and concurs that the changes do correct the TSs to the appropriate test value. Therefore, the staff finds the changes acceptable.

ITS 3.1.5 CTS LCO 3/4.1.3,"Movable Control Assemblies," Action b, reduce distance from 19 inches to 9.9 inches as criterion for misalignment among CEAs and to open the RTCBs on exceeding this criterion.

The proposed change is to reduce the misalignment-of-CEA criterion in Action b of CTS 3.1.3, "Movable Control Assemblies," from 19 inches to 9.9 inches so that this criterion in ITS 3.1.5 would be the 9.9 inches. Also, with the misalignment criterion being exceeded, Action b of CTS 3.1.3 states that the unit would be "in at least Hot Standby within 6 hours" whereas the Action F for ITS 3.1.5 would be to "open the reactor trip breakers." Therefore, the proposed ITS 3.1.5 Action F requires, that with more than one full-length or part-length CEA misaligned from any other CEA in its group by more than 9.9 inches, that the reactor trip beakers are opened.

This proposed TS change would affect the action to be taken for a multiple misaligned CEA event and the related Bases. This proposed action is more restrictive than the CE Standard Technical Specifications (CE-STS) and the CTS. The plant-specific analysis for multiple misaligned CEAs indicates that two or more CEAs misaligned more than 9.9 inches could result in a situation outside the design basis. The appropriate action for multiple CEA misalignments is to open the trip breakers immediately as the proposed Action F indicates. The fact that the CTS were nonconservative with respect to multiple misaligned CEAs was discovered during the PVNGS corrective actions program in January 1989. Since that time, the licensee has had Procedure 40A09ZZ11 in place which requires immediately opening the trip breakers for multiple misaligned CEAs. This is the same action as is required by the proposed Action F. The proposed change to the Bases for ITS 3.1.5 provides an appropriate explanation.

The proposed Action F is more conservative than the CE-STS and is needed because of the results of the plant-specific analysis for the multiple misaligned CEAs. Therefore, the proposed change to Action F of ITS 3.1.5, and the change to the Bases, are acceptable.

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ITS 3.8.2 CTS LCO 3/4.8.2,"Electrical Power Systems - DC Sources," replacement of existing round cells with low specific gravity cells manufactured by vendors not specified in CTS.

By the letter dated December 17, 1997, the licensee requested amendments to the CTS 3/4.8.2, "Electrical Power System, DC Sources," for PVNGS. The proposed CTS 3/4.8.2 amendments would allow replacement of the existing AT&T round cell, Class 1E, 125 volt dc batteries with low specific gravity cell batteries that are manufactured by vendors other than those specified in the current specifications. This proposed amendments will replace the reference to Exide with a generic reference to "low specific gravity cells" and delete the asterisked footnotes referring to one-time exemptions that are in the subject Technical Specifications for Units 2 and 3. This amendments is necessary because the licensee plans to replace the existing high specific gravity round cell batteries with conventional low specific gravity rectangular cell batteries during the next refueling outage of each unit. The batteries are being replaced as a preventive measure due to recent industry and PVNGS experience with the high specific gravity round cell batteries. The high specific gravity round cell batteries are experiencing premature capacity loss.

The Class 1E, 125V dc power system at PVNGS consists of two independent and redundant electrical power subsystems (Trains A and B). Each train consists of two 125V dc channels (Channels A and C for Train A, Channels B and D for Train B). Each 125V dc channel is formed by one battery bank, one battery charger, one inverter, and the associated distribution system. Each train of 125V dc power is provided with a backup battery charger that may be manually connected to either dc channel in that train. The backup battery charger is configured such that it may be connected to only one dc channel in its respective train at a time.

The proposed CTS 3/4.8.2 amendments would allow replacement of the existing AT&T round cell, Class 1E, 125 Volt dc cells with low specific gravity cells that are manufactured by vendors other than those specified in the current specifications. The proposed amendments will replace the reference to Exide with a generic reference to "low specific gravity cells" and delete the asterisked footnotes referring to one-time exemptions that are in the subject Technical Specifications for Units 2 and 3.

The licensee proposed to revise Technical Specification 3/4.8.2 and Bases Section 3/4.8 to replace all references to Exide with generic references of "low specific gravity cells." The licensee also proposes to delete the asterisked footnote on page 3/4 8-10 of the Unit 2 and 3 Technical Specifications. The asterisked footnote on page 3/4 8-10 of the Unit 2 Technical Specifications was requested on a one-time basis to allow refueling operations to continue during the sixth refueling outage with Class 1E, 125V dc batteries that did not conform to the battery capacity specification. This footnote expired when Unit 2 entered Mode 4 on coming out of the sixth refueling outage. The asterisked footnote on page 3/4 8-10 of the Unit 3 CTS provided a one-time deferral of the battery service test to allow Unit 3 to complete that cycle of operation without shutting down to perform this required surveillance test. Since this footnote is no longer required, the licensee proposed to delete it from the Unit 3 CTS.

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The proposed CTS 3/4.8.2 amendments is necessary because the licensee plans to replace the existing high specific gravity round cell batteries with conventional low specific gravity rectangular cell batteries during the next refueling outage for each unit. The batteries are being replaced as a preventive measure due to recent industry and PVNGS experience with the high specific gravity round cell batteries. The high specific gravity round cell batteries are experiencing premature capacity loss.

The specifications for both the low specific gravity rectangular cell and the high specific gravity round cell batteries will be maintained in the Technical Specifications to allow plant operation with both types of batteries during the replacement sequence of the battery banks. This flexibility is required so that one train of Class 1E, 125V dc power sources is maintained operable during the replacement of the battery banks of the other train. Once replacement of all of the high specific gravity round cell batteries has been completed in all three units, the licensee will submit a followup technical specification amendments request to remove the reference to the high specific gravity round cell batteries from The ITS for each unit.

The proposed CTS 3/4.8.2 amendments would allow the licensee to replace the existing high specific gravity round cell batteries with cells from manufacturers other than Exide. Replacing the high specific gravity round cell batteries, which have experienced premature capacity loss, with low specific gravity rectangular cell batteries that have extensive industry experience can enhance the overall reliability and performance of the Class 1E, 125V dc systems. The proposed amendments does not change the performance criteria or cell parameters for the Class 1E 125V dc sources that are defined in the current Technical Specifications for each unit. The proposed amendments therefore only replaces the reference to Exide with a more generic reference to "low specific gravity cells," and deletes asterisked footnotes referring to one-time exemptions that are in the CTS for Units 2 and 3. It is concluded that the amendments involve essentially editorial changes.

Based on the above, the NRC staff concludes that the proposed CTS 3/4.8.2 amendments are acceptable. [Note: This proposed change was noticed in 63 FR 2272 dated 01/14/98, and is TAC Numbers MA0298/9/10. These changes will be approved before the ITS conversion is approved.]

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the ITS conversion amendments for the PVNGS units. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on March XX, 1998 (63 FR XXXXX) for the proposed conversion from the CTS to the ITS for PVNGS. Accordingly, based upon the

environmental assessment, the Commission has determined that issuance of these amendments will not have a significant effect on the quality of the human environment.

Included in these amendments is a change to CTS LCO 3/4.1.3, "Movable Control Assemblies," Action b, to reduce the distance from 19 inches to 9.9 inches as criterion for misalignment among CEAs and to open the RTCBs upon exceeding this criterion. This part of these amendments changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR XXXXX). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments to change CTS LCO 3/4.1.3, "Movable Control Assemblies."

6.0 CONCLUSION

The PVNGS ITS provide clearer, more readily understandable requirements to ensure safe operation of the plants. The NRC staff concludes that the ITS satisfy the guidance in the Commission's Final Policy Statement with regard to the content of TS, and conform to the ISTS provided in NUREG-1432 with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the ITS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36, and other applicable standards. On this basis, the NRC staff concludes that the proposed ITS are acceptable.

The staff has also reviewed the plant-specific changes to the CTS as described in this Safety Evaluation. On the basis of the evaluations described herein for each of the changes, the NRC staff concludes that these changes are acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendments will not be inimical to the common defense and security, or to the health and safety of the public.

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Table of PVNGS Administrative Changes (A)

Discussion of Change		Description	CTS Section	ITS Section
1.0	A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability.	Section 1.0	Section 1.0
1.0	A.2	Clarifies the definition of CHANNEL CALIBRATION by directing testing of required interlocks and displays.	Definitions CHANNEL CALIBRATION	Definitions CHANNEL CALIBRATION
		Clarifies the definition of CHANNEL FUNCTIONAL TEST by directing testing of required interlocks and displays.	Definitions CHANNEL FUNCTIONAL TEST	Definitions CHANNEL FUNCTIONAL TEST
1.0	A.3	Provides a more accurate description of CHANNEL FUNCTIONAL TEST for Digital Computer Channels than exists in the CTS by more accurately describing what is really performed.	Definitions CHANNEL FUNCTIONAL TEST	Definitions CHANNEL FUNCTIONAL TEST
1.0	A.4	Deletes the CTS definition of CONTAINMENT INTEGRITY because the term is not used in ITS.	Definitions CONTAINMENT INTEGRITY	Definitions
1.0	A.5	Deletes the word "conservative" from the definition of CORE ALTERATIONS to avoid potential confusion since there is no reference to what "conservative" means as it relates to CORE ALTERATIONS.	Definitions CORE ALTERATIONS	Definitions CORE ALTERATIONS
1.0	A.6	Deletes the definition of Frequency Notation which is no longer required since ITS lists the specific frequencies in the SRs.	Definitions FREQUENCY NOTATION	Definitions

Discussion of Change		Description	CTS Section	ITS Section
1.0	A.6	Deletes CTS Table 1.1 which lists the Frequency Notations. These items are no longer required since ITS lists the specific frequencies in the SRs.	Section 1.0 Table 1.1	Section 1.0
1.0	A.7	Deletes several CTS Definitions because they are not used in either the LCOs or SRs: GASEOUS RADWASTE SYSTEM, MEMBER(S) OF THE PUBLIC, PURGE - PURGING, REPORTABLE EVENT, SITE BOUNDARY, SOFTWARE, SOURCE CHECK, UNRESTRICTED AREA, VENTILATION EXHAUST TREATMENT SYSTEM, VENTING	Definitions	Definitions
1.0	A.8	Combines three CTS definitions (IDENTIFIED LEAKAGE, PRESSURE BOUNDARY LEAKAGE, and UNIDENTIFIED LEAKAGE) into one compound ITS definition called LEAKAGE.	Definitions	Definitions LEAKAGE
1.0	A.9	Changes the definition of REFUELING by deleting a portion of the CTS definition, relocating some of the content and adding a new segment. This change removes possible confusion or ambiguity associated with the term	Definitions REFUELING	Definitions REFUELING
1.0	A.10	Changes the definition of the word OPERABLE to clarify that the term addresses safety functions and does not encompass non-safety functions that the system may also perform.	Definitions OPERABLE	Definitions OPERABLE
1.0	A.11	Revises the definition of STAGGERED TEST BASIS from dividing the number of systems, components, etc. into the interval, to multiplying the number of systems, components, etc. by the interval to determine the Surveillance Frequency.	Definitions STAGGERED TEST BASIS	Definitions STAGGERED TEST BASIS
1.0	A.12	Not Used	N/A	N/A
1.0	A.13	Adds a footnote to MODES 4 and 5 in CTS Table 1.2 that requires all reactor head closure bolts to be fully tensioned.	Section 1.0 Table 1.2	Section 1.0 Table 1.1-1 Note b
1.0	A.14	Adds three new sections to Technical Specifications to aid in the understanding and use of the new format and presentation style of ITS: 1.2 - Logical Connectors, 1.3 - Completion Times and 1.4 - Frequency.	Section 1.0	Sections 1.2, 1.3 and 1.4

Discussion of Change		Description	CTS Section	ITS Section
1.0	A.15	Modifies the CTS definition of DOSE EQUIVALENT I-131 to reference ICRP-30, Supplement to Part 1, page 192-212, per letter 102-03717-WLS/AKK/NLT, dated June 17, 1996. This change is characterized as administrative because the change reflects the approved TS Amendments 109, 101 and 81 to Units 1, 2, and 3, respectively.	Definitions DOSE EQUIVALENT I-131	Definitions DOSE EQUIVALENT I-131
2.0	A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability.	Section 2.0	Section 2.0
2.0	A.2	Requirements of CTS 6.7.1 are moved to ITS 2.2, SL Violation.	Section 6.7.1	Section 2.2
2.0	A.3	Adds a reference to 10 CFR 50.72, which provides notification requirements.	Section 6.7.1.a	Section 2.2.3
2.0	A.4	Deletes specification of the details to be included in a required Safety Limit Violation Report and instead references 10 CFR 50.73 which provides the details of the content of the required LER.	Sections 6.7.1.b 6.7.1.c	Section 2.2.5
3.0	A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability.	Section 3.0	Section 3.0
3.0	A.2	Moves the phrase "upon failure to meet the LCO..." to LCO 3.0.2 and replaces it with "...as provided in LCO 3.0.2 and LCO 3.0.7."	LCO 3.0.1	LCO 3.0.2
3.0	A.2	Adds LCO 3.0.7 to provide clarification on the use of Special Test Exception (STE) LCOs where applicable.	Section 3.0	LCO 3.0.7
3.0	A.3	Eliminates the details of definition of Noncompliance with a Specification.	LCO 3.0.2	LCO 3.0.2
3.0	A.4	Provides additional clarification that an Action may also be exited if the LCO is no longer applicable.	LCO 3.0.2	LCO 3.0.2
		Requires that Actions be complete if so stated in the individual Specification.	LCO 3.0.2	LCO 3.0.2
3.0	A.5	In the matter of initiating a required shutdown, ITS provides clarification by specifying when the LCO is applicable instead of the CTS approach which provides an exception to specify when the LCO is not applicable.	LCO 3.0.3	LCO 3.0.3

Discussion of Change	Description	CTS Section	ITS Section
3.0 A.5	In the matter of initiating a required shutdown, ITS changes the time required to be in MODES 3 and 5 to be consecutive total time.	LCO 3.0.3	LCO 3.0.3
3.0 A.6	ITS explicitly states that the shutdown is not required to be completed if required actions are completed that allow operation to continue under an LCO or if the LCO conditions are met.	LCO 3.0.3	LCO 3.0.3
	Does not specify that the Completion Time for actions taken to exit 3.0.3 are measured from the time of failure to meet the LCO (this information is detailed in ITS 1.3).	LCO 3.0.3	LCO 3.0.3
3.0 A.7	ITS explicitly allows MODE changes as part of a shutdown of the unit in addition to MODE changes to comply with LCO Actions.	LCO 3.0.4	LCO 3.0.4
3.0 A.8	Not Used	N/A	N/A
3.0 A.9	Provides clarification that failure to meet the specified SRs whether the failure is experienced during the performance of the Surveillance or between performances of the Surveillance constitutes failure to meet the LCO.	LCO 4.0.1	SR 3.0.1
3.0 A.10	Defines the start of the Surveillance Interval as the previous performance or the time a specified condition of the Frequency is met and provides clarification that exceptions to the 25% extension allowance are stated in the individual Specifications.	LCO 4.0.2	SR 3.0.2
3.0 A.11	Provides clarification to the intent of the CTS by allowing MODE changes as part of a shutdown.	LCO 4.0.4	SR 3.0.4
3.0 A.12	Changes the time at which the LCO is declared not met from immediately (with a 24 hours allowance to comply with the Action Requirements) to allowing 24 hours to perform the Surveillances before declaring the LCO not met.	LCO 4.0.4	SR 3.0.4
3.1.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability.	LCO 3.1.1.1	LCO 3.1.1
3.1.1 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.1.1 Note *	LCO 3.1.1
3.1.1 A.3	Changes the required time to initiate boration when the SHUTDOWN MARGIN is less than the value in the COLR from "immediately" to "within 15 minutes."	LCO 3.1.1.1 Action	LCO 3.1.1 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.1.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability.	LCO 3.1.1.2	LCO 3.1.2
3.1.2 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.1.2 Note *	LCO 3.1.2
3.1.2 A.3	Changes the required time to initiate boration when the SHUTDOWN MARGIN is less than the value in the COLR from "immediately" to "within 15 minutes."	LCO 3.1.1.2 Action a	LCO 3.1.2 Action A
3.1.2 A.4	Changes the required time to vary CEA positions and/or initiate boration when K_{N-1} is greater than or equal to .99 from "immediately" to "within 15 minutes."	LCO 3.1.1.2 Action b	LCO 3.1.2 Action B
3.1.2 A.5	Deletes the "or" from "and/or" in regarding to initiating boration when K_{N-1} is greater than or equal to .99.	LCO 3.1.1.2 Action b	LCO 3.1.2 Action B
3.1.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.1.2	LCO 3.1.3
3.1.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.1.3	LCO 3.1.4
3.1.4 A.2	Removes cross-reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.1.3 Note *	LCO 3.1.4
3.1.4 A.3	Modifies the Applicability of SR 3.1.4.1 and 3.1.4.2 by the use of a NOTE which states that the SRs are not required to be performed prior to MODE 2 entry ($K_{eff} \geq .99$).	LCO 3.1.1.3	SR 3.1.4.1 SR 3.1.4.2
3.1.4 A.4	Expands the discussion regarding extrapolation and/or compensation of MTC measured values.	SR 4.1.1.3.1	SR 3.1.4.2 NOTE 2
3.1.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.3.1 LCO 3.1.3.2	LCO 3.1.5
3.1.5 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.3.1 Note *	LCO 3.1.5

Discussion of Change	Description	CTS Section	ITS Section
3.1.5 A.3	Removes cross reference note to 3.1.3.1, 3.1.3.5, 3.1.3.6 and 3.1.3.7 because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.3.2 Action c	LCO 3.1.5
3.1.5 A.4	Deletes the Action requirement addressing drop time exceeding the limit and adds a Surveillance to verify drop time "prior to reactor criticality" that effectively prevents entering the MODES in which the deleted Action applied (MODE 1 and 2).	LCO 3.1.3.4 Action a	SR 3.1.5.5
3.1.5 A.5	Deletes separate Action to address the situation where a full-length CEA is inoperable and trippable since the pertinent requirements are addressed in ITS actions applicable to all CEA's.	LCO 3.1.3.1 Action d	LCO 3.1.5 LCO 3.1.7
3.1.5 A.6	Deletes separate Action to address the situation where a part length CEA is inoperable and inserted in the core since the pertinent requirements are addressed in ITS sections.	LCO 3.1.3.1 Action e	LCO 3.1.5 LCO 3.1.8
3.1.5 A.7	Cross-reference of CTS 3.1.3.6 and 3.1.3.7 is removed since it is not necessary or used in the ITS.	3.1.3.1	3.1.5
3.1.5 A.8	Additional descriptive language is added for CEA misalignment to ITS 3.1.5 Action A.	3.1.3.1	3.1.5
3.1.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.3.5	LCO 3.1.6
3.1.6 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.3.5 Note *	LCO 3.1.6
3.1.6 A.3	Changes the MODE applicability from MODES 1 and 2 with $K_{eff} \geq 1.0$ to simply MODES 1 and 2 with any CEA not fully withdrawn.	LCO 3.1.3.5	LCO 3.1.3.6
3.1.6 A.4	Removes the requirement to determine that each shutdown CEA is withdrawn within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to criticality since it is still required by LCO 3.0.4.	SR 4.1.3.5	LCO 3.1.6
3.1.6 A.5	Deletes a cross reference for an inoperable CEA to another LCO and replaces it with the same Action that requires that the plant be in MODE 3 within 6 hours.	LCO 3.1.3.5 Action	LCO 3.1.6 Action B

Discussion of Change	Description	CTS Section	ITS Section
3.1.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.3.6	LCO 3.1.7
3.1.7 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.3.6 Note *	LCO 3.1.7
3.1.7 A.3	Changes the MODE applicability requirements and adds a NOTE to SR 3.1.7.1 that states that the SR is not required to be performed prior to MODE 2 entry which maintains the CTS requirements.	LCO 3.1.3.6	SR 3.1.7.1
3.1.8 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.3.7	LCO 3.1.8
3.1.8 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.1.3.7 Note *	LCO 3.1.8
3.1.8 A.3	Deletes the reference to part-length CEA groups be maintained within limits "with COLSS in service or out of service" since the insertion limits do not change based on the service status.	LCO 3.1.3.7	LCO 3.1.8
3.1.8 A.4	Deletion of the reference to CTS 4.1.3.1.2 since it is not necessary to exempt condition entry to perform the SR.	3.1.3.7	3.1.8
3.1.9 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.10.1	LCO 3.1.9
3.1.9 A.2	Changes the required time to initiate boration from "immediately" to "within 15 minutes."	LCO 3.10.1 Actions a, b	3.1.9 Action A
3.1.9 A.3	Changes the reference to requirements of SDM Specifications (3.1.1.2) to reflect its relocation (3.1.6 and 3.1.7).	LCO 3.10.1	LCO 3.1.9
3.1.9 A.4	Deletion of the reference to shutdown margin, it is not needed since it is based on CEA position and not applicable to this STE.	3.10.1	3.1.9
3.1.10A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.10.2	LCO 3.10.1

Discussion of Change	Description	CTS Section	ITS Section
3.1.11A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.10.4	LCO 3.11.1
3.1.11A.2	Adds specification of a Completion Time (15 minutes) for the Action to return LHR and DNBR to within Limits.	LCO 3.10.4 Action	LCO 3.1.11
3.2.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.1	LCO 3.2.1
3.2.1 A.2	Not Used	N/A	N/A
3.2.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.2	LCO 3.2.2
3.2.2 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.2.2	LCO 3.2.2
3.2.2 A.3	Deletes the exception to Specification 4.0.4 since the surveillance is required once after each fuel load with thermal power greater than 40% and not required to be performed prior to entry into the applicability.	4.2.2.1	3.2.2
3.2.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.3	LCO 3.2.3
3.2.3 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.2.3	LCO 3.2.3
3.2.3 A.3	Combines the 2 hours allowed to verify that the AZIMUTHAL POWER TILT is within its limits with the additional 2 hours allowed to reduce THERMAL POWER to less than 50% if the verification is not completed as required into a single 4 hour requirement to reduce power.	LCO 3.2.3 Action b.2	LCO 3.5.3 Action B.1
3.2.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.4	LCO 3.2.4

Discussion of Change	Description	CTS Section	ITS Section
3.2.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.7	LCO 3.2.5
3.2.5 A.2	Removes cross reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.2.7	LCO 3.2.5
3.3.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.1	LCO 3.3.1
3.3.1 A.2	Rewords the Surveillance Interval to use the ITS STAGGERED TEST BASIS but does not change the interval.	SR 4.3.1.3	SR 3.3.1.13
3.3.1 A.3	Rewords the Surveillance Requirement to use the term CHANNEL FUNCTIONAL TEST to capture the concept of the CTS terminology "logic for the bypasses" with equivalent intent.	SR 4.3.1.2	SR 3.3.1.12
3.3.1 A.4	The LCO specifies the number of RPS trip and bypass removal channels required to be OPERABLE for each function in the associated table. CTS didn't include the number of channels required in the LCO statement because of the mix of equipment included in the CTS Table.	LCO 3.3.1	LCO 3.3.1
3.3.1 A.5	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each RPS Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.1	LCO 3.3.1 Actions NOTE
3.3.1 A.6	Rewords the statement regarding the number of inoperable automatic RPS trip channels compared to the total number of channels that require Action entry but does not change the requirement.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.1 Action A
3.3.1 A.7	Rewords the statement regarding the number of inoperable automatic RPS trip channels compared to the minimum number of channels that require Action entry but does not change the requirement.	LCO 3.3.1 Table 3.3-1 Action 3	LCO 3.3.1 Action B
3.3.1 A.8	Removes the CHANNEL FUNCTIONAL TEST cross-reference from the table NOTE addressing the limited calibration to adjust instrumentation to agree with the calorimetric calculation since the calibration required is described in the SR.	LCO 3.3.1. Table 4.3-1 Note (2)	SR 3.3.1.4

Discussion of Change	Description	CTS Section	ITS Section
3.3.1 A.9	Removes the words "steady state" from the phrase "Verify the total steady state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS flowrate" because flow is always "steady state" under the test performance conditions.	LCO 3.3.1 Table 4.3-1 Note (7)&(8)	SR 3.3.1.2 SR 3.3.1.5
3.3.1 A.10	Removes the unneeded words "quarterly" and "current" from the phrase "The quarterly channel functional test shall include verification that the correct current values of addressable constants are installed in each operable CPC. Quarterly is the standard ITS frequency.	LCO 3.3.1 Table 4.3-1 Note (9)	SR 3.3.1.7
3.3.1 A.11	Adds a new Action that requires entry into MODE 3 within 6 hours if the Completion Time for the prior Actions is not met instead of forcing entry into LCO 3.0.3. Both CTS and ITS require reports per 10CFR. ITS requires reports per 10CFR50.73, whereas, CTS requires reports per 10CFR50.72 and 10CFR50.73.	LCO 3.3.1	LCO 3.3.1 Action G
3.3.1 A.12	Eliminates reference to the RPS Logic and Actuation devices and the statement that the trip setpoint and setpoint allowable values are not applicable to the equipment. The format of the ITS makes this unnecessary.	Section 2.2 Table 2.2-1 Items II, III	LCO 3.3.1
3.3.1 A.13	Eliminates information related to low pressurizer pressure setpoints that is not necessary because it applies to a MODE in which the functions are not required to be OPERABLE.	Section 2.2 Table 2.2-1 Notation (2) LCO 3.3.1 Table 3.3-1 Notation (b)	LCO 3.3.1
3.3.1 A.14	Changes the units of reference for the Reactor Coolant Flow-Low Rate Allowable Value from "psi/sec" to "psid/sec" with no change to the time dependent functional values.	Section 2.2 Item 1.A.7.a	LCO 3.3.1 Table 3.3.1-1 Functions 12, 13

Discussion of Change	Description	CTS Section	ITS Section
3.3.1 A.15	Eliminates reference to the Core Protection Calculators since it is not a trip function and its outputs are included in the table.	LCO 3.3.1 Table 4.3-1	LCO 3.3.1 Table 3.3.1-1 Functions 14, 15
3.3.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.1 LCO 3.3.2	LCO 3.3.2
3.3.2 A.2	The LCO specifies the number of reactor protective instrumentation channels and bypasses required to be OPERABLE for each function in the associated table. CTS didn't include the number of channels required in the LCO statement because of the mix of equipment included in the CTS Table.	LCO 3.3.1	LCO 3.3.2
3.3.2 A.3	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each RPS Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.1	LCO 3.3.2 NOTE
3.3.2 A.4	Rewords the Surveillance Frequency of the Reactor Trip RESPONSE TIME Surveillance to use the ITS STAGGERED TEST BASIS definition. The actual interval is not changed.	SR 4.3.1.3	SR 3.3.2.5
3.3.2 A.5	Rewords the Surveillance Requirement to use the term CHANNEL FUNCTIONAL TEST to capture the concept of the CTS terminology "logic for the bypasses" with equivalent intent.	SR 4.3.1.2	SR 3.3.2.3
3.3.2 A.6	Clarifies the intent of the breaker and CEA portion of the note regarding the protective system trip breakers in the closed position, CEA drive system capable of CEA withdrawal and fuel in the reactor vessel. Removes the reference to fuel in the reactor vessel as unnecessary due to the MODE definition in the ITS.	LCO 3.3.1 Table 3.3-1 Note (*)	LCO 3.3.2 Table 3.3.2-1 Note (a)
3.3.2 A.7	Rewords the statement regarding the number of inoperable automatic RPS trip channels compared to the total number of channels that require Action entry but does not change the requirement.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.2 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.3.2 A.8	Rewords the statement regarding the number of inoperable automatic RPS trip channels compared to the minimum number of channels that require Action entry but does not change the requirement.	LCO 3.3.1 Table 3.3-1 Action 3	LCO 3.3.2 Action B
3.3.2 A.9	Eliminates the NOTE reference to MODES since ITS provides this information elsewhere and CTS did not. Changes the reference to "setpoint value" by deleting the word "value."	Section 2.2 Table 2.2-1 Note (3)	LCO 3.3.2 Table 3.3.2-1 Note (b)
3.3.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.1 LCO 3.1.3.6	LCO 3.3.3
3.3.3 A.2	Rewords the statement regarding the number of CEAC channels required to be OPERABLE as a result of the restructuring of ITS but does not change the content.	LCO 3.3.1	LCO 3.3.3
3.3.3 A.3	Rewords the SR by replacing "Core Protection Calculators" with "CEACs" to clarify that the phrase applies to both CPCs and CEACs.	SR 4.3.1.5	LCO 3.3.3 Action D
3.3.3 A.4	Simplifies the wording regarding withdrawal of CEA groups to say "fully withdrawn and maintained full withdrawn" in place of referencing other Specifications.	LCO 3.3.1 Table 3.3-1 Action 6.b.2.a 6.b.2.c	LCO 3.3.3 Action B.2
3.3.3 A.5	Moves the SR requirement to increase the surveillance frequency on CEA alignment checks from the CEA Alignment SR to the CEAC LCO, the actual frequency remains unchanged.	LCO 3.3.1 Table 3.3-1 Action 6.a SR 4.1.3.1.1	LCO 3.3.3 Action A
3.3.3 A.6	Adds an Action that requires entry into MODE 3 within 6 hours if the other Completion Times are not met instead of forcing entry into LCO 3.0.3. Both CTS and ITS require reports per 10CFR. ITS requires reports per 10CFR50.73, whereas, CTS requires reports per 10CFR50.72 and 10CFR50.73.	LCO 3.3.1	LCO 3.3.3 Action E

Discussion of Change	Description	CTS Section	ITS Section
3.3.3 A.7	Removes cross-reference note to Special Test Exceptions because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.3.1 Table 3.3-1 Note (a)	LCO 3.3.3
3.3.3 A.8	Eliminates reference to specific sub-sections regarding DNBR margin establishment since the referenced ITS section provides the necessary detail and sub-section reference is not required.	LCO 3.3.1 Table 3.3-1 Action 6.b.1	LCO 3.3.3 Action B
3.3.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.1	LCO 3.3.4
3.3.4 A.2	CTS LCO Applicability refers to a Table, ITS will not use a table format. Changes Applicability wording to clarify that the intent is "With any Reactor trip circuit Breakers (RTCBs) in the closed position and any CEA capable of withdrawal" and deletes the words "with fuel in the vessel" since the ITS definition of MODE makes them unnecessary.	LCO 3.3.1	LCO 3.3.4
3.3.4 A.3	The LCO specifies the number of reactor protective instrumentation channels and bypasses required to be OPERABLE for each function in the associated table. CTS didn't include the number of channels required in the LCO statement because of the mix of equipment included in the CTS Table.	LCO 3.3.1	LCO 3.3.4
3.3.4 A.4	Adds an Action that requires entry into MODE 3 within 6 hours if the other Completion Times are not met and therefor avoids the LCO 3.0.3 entry required by CTS which contains no specific Action to cover this condition.	LCO 3.3.1	LCO 3.3.4 Action E
3.3.4 A.5	Rewords the Action statement regarding the required number of OPERABLE channels to the perspective of the number of inoperable channels but does not change the requirements.	LCO 3.3.1 Actions	LCO 3.3.4 Action A
3.3.4 A.6	Rewords the Action statement regarding the conditions under which an inoperable RTCB may be closed to state the name of the Surveillance Test (CHANNEL FUNCTIONAL TEST) instead of referencing the SR number.	LCO 3.3.1 Table 3.3-1 Action 5	LCO 3.3.4 Action B

Discussion of Change	Description	CTS Section	ITS Section
3.3.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.2	LCO 3.3.5
3.3.5 A.2	Changes the wording of the ESFAS Instrumentation LCO to specify the number of OPERABLE channels in the LCO instead of using a table for this purpose.	LCO 3.3.2	LCO 3.3.5
3.3.5 A.3	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each ESFAS Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.2 Actions	LCO 3.3.5 Actions NOTE
3.3.5 A.4	Rewords the Surveillance Frequency of the ENGINEERED SAFETY FEATURES response time Surveillance to use the ITS STAGGERED TEST BASIS definition. The actual interval is not changed.	SR 4.3.2.3	SR 3.3.5.4
3.3.5 A.5	Rewords the statement regarding the number of inoperable ESFAS channels compared to the total number of channels that require Action entry but does not change the requirement.	LCO 3.3.2 Table 3.3-3 Action 13	LCO 3.3.5 Action A
3.3.5 A.6	Rewords the statement regarding the number of inoperable ESFAS channels compared to the minimum number of channels that require Action entry but does not change the requirement.	LCO 3.3.2 Table 3.3-3 Action 14	LCO 3.3.5 Action B
3.3.5 A.7	Adds an Action that requires entry into MODE 3 within 6 hours if the other Completion Times are not met instead of forcing entry into LCO 3.0.3. Both CTS and ITS require reports per 10CFR. ITS requires reports per 10CFR50.73, whereas, CTS requires reports per 10CFR50.72 and 10CFR50.73.	LCO 3.3.2	LCO 3.3.5 Action E
3.3.5 A.8	Rewords notation wording to refer to the setpoint instead of the setpoint value, clarifies that the setpoints increase until the normal value is reached, does not explicitly state that the setpoint reductions are accomplished manually and removes reference to MODES 3 and 4 and consistent with ITS table format.	LCO 3.3.2 Table 3.3-3 Note (a), (b)	LCO 3.3.5 Table 3.3.5-1 Note (a), (b)

Discussion of Change	Description	CTS Section	ITS Section
3.3.5 A.9	Eliminates the LCO 3.0.4 exclusion because operation with a single channel is allowed for an unlimited period of time and ITS LCO 3.0.4 permits MODE changes in this condition.	LCO 3.3.2 Table 3.3-3 Action 13 Note *	LCO 3.3-5 Action A
3.3.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.2	LCO 3.3.6
3.3.6 A.2	Changes the wording of the ESFAS Instrumentation LCO to specify the number of OPERABLE channels in the LCO instead of using a table for this purpose.	LCO 3.3.2	LCO 3.3.6
3.3.6 A.3	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each ESFAS Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.2 Actions	LCO 3.3.6 Actions NOTE
3.3.6 A.4	Rewords the statement regarding the number of inoperable ESFAS channels compared to the total number of channels that require Action entry but does not change the requirement.	LCO 3.3.2 Table 3.3-3 Action 12, 15, 16	LCO 3.3.5 Action A
3.3.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.2	LCO 3.3.7
3.3.7 A.2	Not Used	N/A	N/A
3.3.7 A.3	Includes ESFAS Loss of Voltage and Degraded Voltage functions and uses the terminology "function" instead of "and bypasses" which is inclusive.	LCO 3.3.2	LCO 3.3.7
3.3.7 A.4	Eliminates the use of a table with notes to specify Actions associated with an inoperable channel since the LCO presents singular Actions which apply to both Functions (Loss of Voltage and Degraded Voltage).	LCO 3.3.2 Table 3.3-3	LCO 3.3.7 Actions
3.3.7 A.5	Rewords the statement regarding the number of inoperable LOVS channels compared to the total number of channels that require Action entry but does not change the requirement.	LCO 3.3.2 Table 3.3-3 Action 13	LCO 3.3.7 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.3.7 A.6	Rewords the statement regarding the number of inoperable LOVS channels compared to the minimum number of channels that require Action entry but does not change the requirement.	LCO 3.3.2 Table 3.3-3 Action 14	LCO 3.3.7 Action B
3.3.7 A.7	Eliminates reference to other Actions since the format of ITS makes them unnecessary.	LCO 3.3.2 Table 3.3-3 Action 19 (a), (b)	LCO 3.3.7 Actions
3.3.8 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.3.1	LCO 3.3.8
3.3.8 A.2	ITS Action clarifies the intent of CTS to enter Actions when both radiation monitors (RU-37 and RU-38) are inoperable.	LCO 3.3.3.1 Table 3.3-6 Action 25	LCO 3.3.8 Action A, B
3.3.8 A.3	Rewords the Action for the containment purge valve isolation system inoperable to eliminate the need to refer to a different section of the ITS for information.	LCO 3.3.3.1 Table 3.3-6 Action 25	LCO 3.3.8 Action A, C
3.3.8 A.4	Adds a clarifying NOTE to the Applicability of the LCO which states "Only required when the penetration is not isolated by at least one closed automatic valve, closed manual valve, or blind flange." This note is equivalent to the CTS applicability "When purge system is being used."	LCO 3.3.3.1 Table 3.3-6 NOTE #	LCO 3.3.8 NOTE
3.3.8 A.5	Rewords the Surveillance Frequency requirement for the radiation monitoring channel but does not change the frequency.	SR 4.3.3.1 Table 4.3-3 Inst. 1.C	SR 3.3.8.1
3.3.9 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.2 LCO 3.3.3.1	LCO 3.3.9
3.3.9 A.2	Reworded the LCO to clarify the number of CREFAS channels required for operability.	LCO 3.3.3.1	LCO 3.3.9
3.3.9 A.3	Rewords the Surveillance Frequency of the ENGINEERED SAFETY FEATURES response time (CREFAS) Surveillance to use the ITS STAGGERED TEST BASIS definition. The actual interval is not changed.	SR 4.3.2.3	SR 3.3.9.6

Discussion of Change	Description	CTS Section	ITS Section
3.3.9 A.4	Changes the Action wording to clarify the intent regarding both CREFAS Manual Trip, Actuation Logic or radiation monitor channels inoperable.	LCO 3.2 Table 3.3-3 Action 18	LCO 3.3.9 Action A, C
3.3.10A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.3.1 LCO 3.3.3.6	LCO 3.3.10
3.3.10A.2	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each PAM Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.3.6	LCO 3.3.10 NOTE
3.3.10A.3	Eliminates the Action statement in CTS that directs attention to Table 3.3-10 because it is unneeded due to the format of ITS.	LCO 3.3.3.6 Action a	LCO 3.3.10
3.3.11A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.3.5	LCO 3.3.11
3.3.11A.2	Eliminates reference to a table for the Remote Shutdown System Surveillance Requirements since ITS format does not use a table for this purpose.	SR 4.3.3.5.a	SR 3.3.11.3
3.3.11A.3	Adds a NOTE to the Specification Actions that states that separate condition entries are allowed for each PAM Function. The format of the ITS makes this NOTE necessary where in CTS the situation was clear without this note.	LCO 3.3.3.5	LCO 3.3.11 NOTE
3.3.12A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.2.7 LCO 3.3.1	LCO 3.3.12
3.3.12A.2	Changes the system name in the LCO from "startup channel high neutron flux alarm" to "Boron Dilution Alarm System (BDAS)" and replaces the word "both" with "two" for consistency with NUREG-1432.	LCO 3.1.2.7	LCO 3.3.12

Discussion of Change	Description	CTS Section	ITS Section
3.3.12A.3	Changes the Action Completion time to determine the RCS boron concentration from "when entering MODE 3, 4, 5 or 6, or at the time the alarm is determined to be inoperable" to "immediately" as a clarification to boron concentration not met during specified applicability.	LCO 3.1.2.7 Action a.1	LCO 3.3.12 Action A
3.3.12A.4	Changes the Action Completion time to determine the RCS boron concentration from "when entering MODE 3, 4, 5 or at the time both alarms are determined to be inoperable" to "immediately" as a clarification.	LCO 3.1.2.7 Action b.1	LCO 3.3.12 Action B
3.3.12A.5	Rewords the Surveillance Requirement Frequency for BDAS CHANNEL CHECKS to simplify it and eliminate the need for two intervals. The frequency of the SR is unchanged.	SR 4.1.2.7	SR 3.3.12.1
3.4.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.2.5 LCO 3.2.6 LCO 3.2.8	LCO 3.4.1
3.4.1 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.2.6	LCO 3.4.1
3.4.1 A.3	Not Used	N/A	N/A
3.4.1 A.4	Modified the Pressurizer Pressure LCO to reflect the more restrictive limits specified in a proposed PVNGS license amendment (Letter 102-03717 dated June 17, 1996). This change is characterized as administrative because the change reflects approved TS Amendments 109, 101 and 81 to Units 1, 2, and 3, respectively.	LCO 3.2.8	LCO 3.4.1
3.4.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.1.1.4	LCO 3.4.2
3.4.2 A.2	Modifies the Action Completion time allowed to return RCS Cold Leg Temperature to $\geq$ 545°F and reduce power to enter MODE 3 from 15 minutes and 15 minutes respectively to a total of 30 minutes without specification of sub-segments.	LCO 3.1.1.4 Action	3.4.2 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.4.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.8.1	LCO 3.4.3
3.4.3 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.4.8.1	LCO 3.4.3
3.4.3 A.3	Deletes the statement that a determination that the RCS remains acceptable for continued operations is required since it is implicit in the ITS Actions.	LCO 3.4.8.1 Action	LCO 3.4.3 Action A, C
3.4.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.1.1	LCO 3.4.4
3.4.4 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.4.1.1	LCO 3.4.4
3.4.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.1.2	LCO 3.4.5
3.4.5 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.4.1.2	LCO 3.4.5
3.4.5 A.3	Describes the condition that defines inoperable, not in operation, or condition not met (one RCS loop inoperable) and splits the CTS Action into two separate Actions.	LCO 3.4.1.2 Action a	LCO 3.4.5 Action A, C
3.4.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.1.3	LCO 3.4.6
3.4.6 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.4.1.3	LCO 3.4.6
3.4.6 A.3	Adds the statement "...no RCS loop or SDC train OPERABLE" to clarify the intent of "with no reactor coolant or shutdown cooling loop in operation."	LCO 3.4.1.3 Action b	LCO 3.4.6 Action C
3.4.6 A.4	Describes the condition that defines inoperable or not in operation and splits the CTS Action into two separate Actions.	LCO 3.4.1.3 Action a	LCO 3.4.6 Action A, B

Discussion of Change	Description	CTS Section	ITS Section
3.4.6 A.5	Changes the word "dilution" to "reduction" in the footnote discussion regarding prohibiting dilution of RCS boron concentration during the time when all RCPs and SDC pumps are de-energized.	LCO 3.4.1.3 Footnote *	LCO 3.4.6 NOTE 1
3.4.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.1.4.1	LCO 3.4.7
3.4.7 A.2	Changes the word "dilution" to "reduction" in the footnote discussion regarding prohibiting dilution of RCS boron concentration during the time when all RCPs and SDC pumps are de-energized.	LCO 3.4.1.4.1 Footnote *	LCO 3.4.7 NOTE 1
3.4.7 A.3	Adds the words "required SDC train inoperable" to "with no shutdown cooling loop in operation...." to clarify the intent of the Action.	LCO 3.4.1.4.1 Action b	LCO 3.4.7 Action B
3.4.7 A.4	Describes the condition that defines inoperable or not in operation and splits the CTS Action into two separate Actions.	LCO 3.4.1.4.1 Footnote *	LCO 3.4.7 NOTE 1
3.4.8 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.1.4.2	LCO 3.4.8
3.4.8 A.2	Changes the word "dilution" to "reduction" in the footnote discussion regarding prohibiting dilution of RCS boron concentration during the time when all RCPs and SDC pumps are de-energized.	LCO 3.4.1.4.2 Footnote *	LCO 3.4.8 NOTE 1.b
3.4.8 A.3	Not Used	N/A	N/A
3.4.9 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.3.1	LCO 3.4.9
3.4.9 A.2	Adds a NOTE regarding the applicability of the pressurizer steady state water level specification to clarify the transient conditions in which the limit doesn't apply.	LCO 3.4.3.1	LCO 3.4.9 NOTE
3.4.10A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.2.2	LCO 3.4.10

Discussion of Change	Description	CTS Section	ITS Section
3.4.10A.2	Changes the manner of reference to Inservice Testing requirements from LCO 4.0.5 to the Inservice Testing Program for the pressurizer safety valves.	SR 4.4.2.2	SR 3.4.10.1
3.4.11A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.2.2	LCO 3.4.11
3.4.11A.2	Changes the manner of reference to Inservice Testing requirements from LCO 4.0.5 to the Inservice Testing Program for the pressurizer safety valves.	SR 4.4.2.1	SR 3.4.11.1
3.4.12A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.10	LCO 3.4.12
3.4.12A.2	Deletes the phrase "from either location" to reflect the removal of the reactor head vents from Technical Specifications which leaves only pressurizer vents in ITS.	LCO 3.4.10 Action a, b	LCO 3.4.12
3.4.12A.3	Deletes the statement that the pressurizer vent valves have to be closed since the requirements for the valves to be closed are contained in other parts of the specifications.	LCO 3.4.10	3.4.12
3.4.13A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.8.3	LCO 3.4.13
3.4.13A.2	Moves clarifying information regarding the circumstances when SDC suction line relief valves must be placed in service into a NOTE.	LCO 3.4.8.3	LCO 3.4.13 NOTE
3.4.13A.3	Deletes the specific statement of LCO 3.0.4 exclusion for MODE 5 and 6 since the ITS definition of LCO 3.0.4 excludes these modes.	LCO 3.4.8.3 Action f	LCO 3.4.13
3.4.13A.4	Rewords and moves the prohibition on starting an RCP if the SG secondary water temperature is more than 100°F greater than any cold leg temperature into a NOTE.	LCO 3.4.8.3 Action a, b, c	LCO 3.4.13 NOTE
3.4.14A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.4 LCO 3.4.5.2	LCO 3.4.14

Discussion of Change	Description	CTS Section	ITS Section
3.4.14A.2	Adds an Action that requires immediate entry into LCO 3.0.3 if one or more Steam Generators is inoperable. This is a reformat of CTS requirements and not a change in requirements.	LCO 3.4.4 Action	LCO 3.4.14 Action C
3.4.14A.3	Eliminates a redundant requirement to monitor the containment atmospheric gaseous and particulate radioactivity monitor at least once per 12 hours.	SR 4.4.5.1.a SR 4.4.5.2.1	SR 3.4.16.1
3.4.14A.4	Eliminates a footnote reference that excludes the provisions of Specification of 4.0.4 and clarifies the related issue requirements for performance such that an explicit exception is not necessary.	SR 4.4.5.2.1.c	SR 3.4.14.1
3.4.15A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.5.2 LCO 3.7.11	LCO 3.4.15
3.4.15A.2	Provides a NOTE which clarifies the intent of the Action and gives explicit instructions for proper application of the Actions to achieve TS compliance.	LCO 3.4.5.2 Action c	LCO 3.4.15 NOTE 1
3.4.16A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.3.3.1 LCO 3.4.5.1	LCO 3.4.16
3.4.16A.2	Removes cross-reference note to another specification because cross-reference notes are not used in ITS or NUREG-1432.	LCO 3.3.3.1 Table 3.3-6 Action 23	LCO 3.4.16
3.4.16A.3	Moves the Action requirements for the Containment Sump Monitoring System to be OPERABLE when the containment atmosphere radioactivity monitor (gaseous and particulate) is inoperable (and vice-versa) to an LCO and Actions.	LCO 3.4.5.1 Action a, b	LCO 3.4.16 LCO 3.4.16 Action C, D
3.4.17A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.4.7	LCO 3.4.17
3.4.17A.2	Relocates the primary coolant specific activity requirements from the LCO to Surveillance Requirements without changing the requirements.	LCO 3.4.7 (a and b)	SR 3.4.17.1 SR 3.4.17.2

Discussion of Change	Description	CTS Section	ITS Section
3.5.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.1	LCO 3.5.1
3.5.1 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.5.1	LCO 3.5.1
3.5.1 A.3	Not Used	N/A	N/A
3.5.1 A.4	Adds an action to require entry into LCO 3.0.3 if two or more SITs are inoperable. This is a clarification to ensure the operators understand the need to enter LCO 3.0.3 in circumstances with multiple SITs inoperable for different Action requirements.	LCO 3.5.1	LCO 3.5.1 Action D
3.5.1 A.5	Removes the requirement regarding the verification that power is removed from the SIT isolation valve operator when pressurizer pressure is above 430 psia. This is administrative since this ITS LCO isn't applicable below 1837 psia.	SR 4.5.1.c	SR 3.5.1.5
3.5.1 A.6	Changes the Action requirement end-states from "HOT SHUTDOWN" to "< 1837 psia" which is an administrative change that reflects the split of this CTS LCO into ITS LCO 3.5.1 and 3.5.2. This Action end-state takes the plant to a condition where LCO 3.5.1 no longer applies.	LCO 3.5.1 Action a, b, c	LCO 3.5.1 Action C
3.5.1 A.7	Reflects the split of this CTS LCO into two ITS LCOs, (3.5.1 and 3.5.2) by changing the MODE of applicability to match the administrative split of requirements.	LCO 3.5.1	LCO 3.5.1
3.5.1 A.8	Not Used	N/A	N/A
3.5.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.1	LCO 3.5.2
3.5.2 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.5.1	LCO 3.5.2
3.5.2 A.3	Not Used	N/A	N/A
3.5.2 A.4	Adds an action to require entry into LCO 3.0.3 if two or more SITs are inoperable. This is a clarification to ensure the operators understand the need to enter LCO 3.0.3 in circumstances with multiple SITs inoperable for different Action requirements.	LCO 3.5.1	LCO 3.5.2 Action D

Discussion of Change	Description	CTS Section	ITS Section
3.5.2 A.5	Reflects the split of this CTS LCO into two ITS LCOs, (3.5.1 and 3.5.2) by changing the MODE of applicability to match the administrative split of requirements.	LCO 3.5.1	LCO 3.5.2
3.5.2 A.6	Not Used	N/A	N/A
3.5.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.2 LCO 3.7.11	LCO 3.5.3
3.5.3 A.2	Relocates the direction to perform specific ECCS SRs "during shutdown" from the Surveillance Requirement to a clarification in the Frequency column.	SR 4.5.2.e	SR 3.5.3.4 SR 3.5.3.5 SR 3.5.3.6
3.5.3 A.3	Changes the wording regarding pumps that must start on upon receipt of an SI actuation signal from "low pressure safety injection pumps" and "high pressure safety injection pumps" to ECCS pumps, which encompasses both LPSI and HPSI pumps.	SR 4.5.2.3.2	SR 3.5.3.5
3.5.3 A.4	Removes cross reference note to a different Specification that contains IST requirements because cross reference notes are not used in ITS or NUREG-1432. The reference is replaced with "In accordance with the Inservice Testing Program."	SR 4.5.2.f	SR 3.5.3.3
3.5.3 A.5	Not Used	N/A	N/A
3.5.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.3	LCO 3.5.4
3.5.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.4	LCO 3.5.5
3.5.5 A.2	Deletes the portion of the figure related to Spent Fuel Pool Volume requirements since this portion of the Specification has been relocated and is therefore no longer applicable.	LCO 3.5.4 Figure 3.1-1	LCO 3.5.5
3.5.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.5.2	LCO 3.5.6

Discussion of Change	Description	CTS Section	ITS Section
3.5.6 A.2	Incorporates the TS change request dated June 28, 1996 which changes the minimum cubic feet of TSP from 464 to 524. This change is characterized as administrative because the change reflects the approved TS Amendments 110, 102 and 82 to Units 1, 2, and 3, respectively.	SR 4.5.2.d.2	SR 3.5.6.1
3.6.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.1.1 LCO 3.6.1.2 LCO 3.6.1.6	LCO 3.6.1
3.6.1 A.2	Combines the use of terms describing aspects of Containment operability (integrity and leakage rates) into the single concept that "Containment shall be OPERABLE." The intent is not changed.	LCO 3.6.1.1 LCO 3.6.1.2 LCO 3.6.1.6	LCO 3.6.1
3.6.1 A.3	Combines the LCO singular focus on "Containment leakage rates) into the single concept that "Containment shall be OPERABLE."	LCO 3.6.1.2	LCO 3.6.1
3.6.1 A.4	Excludes the Containment air locks from the Surveillance for visual examination and leakage rate testing since it is separately addressed in ITS 3.6.2.	SR 4.6.1.2	SR 3.6.1.1
3.6.1 A.5	Deletes specification of requirements related to containment structural integrity since they are now a part of the Containment Tendon Surveillance Program addressed in ITS section 5.5.6.	SR 4.6.1.6.1	SR 3.6.1.2 ITS 5.5.6
3.6.1 A.6	Adds the statement requiring performance of periodic visual examinations of the containment since that examination is required by 10 CFR 50, Appendix J and conformance is controlled by the Containment Leakage Rate Testing Program described in ITS section 5.5.6.	SR 4.6.1.2	SR 3.6.1.1 ITS 5.5.6
3.6.1 A.7	Eliminates a statement requiring conformance to a related Specification since the ITS format does not typically make such cross-references and the referenced Specification's requirements separately require conformance and therefore stand on their own.	SR 4.6.1.1.b	LCO 3.6.1.1

Discussion of Change	Description	CTS Section	ITS Section
3.6.1 A.8	Eliminates an exception to LCO 4.0.2 which allows surveillance intervals to be extended a maximum of 25%. This interval is specified by regulation and the extension does not apply in ITS as stated in the SR 3.0.2 bases and therefore this change is administrative only.	SR 4.6.1.2	SR 3.6.1.1
3.6.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.1.3	LCO 3.6.2
3.6.2 A.2	Eliminates the specific option to restore the inoperable air lock door to OPERABILITY to exit the LCO since it's a basic concept addressed in LCO 3.0.2.	LCO 3.6.1.3 Action	LCO 3.0.2
3.6,2 A.3	Eliminates an unnecessary exception to LCO 3.0.4 since the actions of this LCO permit operation with an inoperable air lock door for an unlimited period of time.	LCO 3.6.1.3 Action a.3	LCO 3.0.4
3.6.2 A.4	Eliminates reference to performing Surveillances in accordance with the Containment Leakage Rate Testing Program "at periodic intervals" since the program specifies the intervals.	SR 4.6.1.3.a	SR 3.6.2.1
3.6.2 A.5	Adds a clarifying NOTE which permits separate Condition entry for each air lock. This was implied in CTS by the wording of the LCO that stated "Each containment air lock shall be OPERABLE."	LCO 3.6.1.3	LCO 3.6.2 NOTE 2
3.6.2 A.6	Adds clarifying NOTES and Actions to remind the licensee that other Specifications may be affected if the leakage rate of the air lock(s) violate Containment Integrity.	LCO 3.6.1.3	LCO 3.6.2 NOTE 3 LCO 3.6.2 Action C.1 SR 3.6.2.1 NOTE
3.6.2 A.7	Adds a clarifying NOTE regarding the applicability of Actions A.1, A.2 and A.3 if Condition C is entered. This NOTE is necessary due to the ITS practice of multiple Condition entry and makes the ITS LCO consistent with the CTS source LCO.	LCO 3.6.1.3 Action b	LCO 3.6.2 Action A NOTE 1

Discussion of Change	Description	CTS Section	ITS Section
3.6.2 A.8	Adds a clarifying NOTE regarding the applicability of Actions B.1, B.2 and B.3 if Condition C is entered.	LCO 3.6.1.3 Action b	LCO 3.6.2 Action B NOTE 1
3.6.2 A.9	Adds a clarifying NOTE which states "An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test."	SR 4.6.1.3	SR 3.6.1.2.1 NOTE 1
3.6.2 A.10	Eliminates an exception to LCO 4.0.2 which allows surveillance intervals to be extended a maximum of 25%. This interval is specified by regulation and the extension does not apply in ITS as stated in the SR 3.0.2 bases and therefore this change is administrative only.	SR 4.6.1.3	SR 3.0.2
3.6.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.1.1 LCO 3.6.1.7 LCO 3.6.3	LCO 3.6.3
3.6.3 A.2	Removes the requirement from SRs for verification of position for valves closed to comply with Actions since these verifications are now required in specific Actions (ITS LCO 3.6.3 Action A.2, C.2 and D.2).	SR 4.6.1.1.a	LCO 3.6.3 ACTIONS
3.6.3 A.3	Adds a NOTE which states "Separate Condition entry is allowed for each penetration flow path" which clarifies the intent of the CTS source LCO.	LCO 3.6.1.7	LCO 3.6.3 NOTE 2
3.6.3 A.4	Rewords the CTS NOTE to provide clarification that Actions are to be complied with for systems made inoperable by inoperable by containment isolation valves.	LCO 3.6.3 Action 1.b, 1.c Note **	LCO 3.6.3 NOTE 3
3.6.3 A.5	Adds a clarifying NOTE to remind the licensee that another Specification may be affected if the leakage rate of a penetration flow path violates Containment Integrity.	LCO 3.6.1.7 LCO 3.6.3	LCO 3.6.3 NOTE 4

Discussion of Change	Description	CTS Section	ITS Section
3.6.3 A.6	Adds clarifying NOTES specifying which Actions are applicable to each type of penetration.	LCO 3.6.1.7 LCO 3.6.3 LCO 3.6.1.1	LCO 3.6.3 Action A, B, C NOTE
3.6.3 A.7	Not Used	N/A	N/A
3.6.3 A.8	Rewords the Action Condition Statement to clarify that they apply to one or more penetration flow paths with inoperable CIVs.	LCO 3.6.1.7 Action a, b, c	LCO 3.6.3 Action A, D
3.6.3 A.9	Eliminates specific reference to restoring operability as a method of exiting the LCO Actions since this is already provided generically in ITS LCO 3.0.2.	LCO 3.6.1.7 Action a, b, c LCO 3.6.3 Action a	LCO 3.0.2
3.6.3 A.10	Eliminates the requirement to perform a 31 day interval position verification Surveillance on a purge valve with leakage exceeding limits in a penetration flow path while in an Action Statement for that condition. That Action separately isolates the valve and requires a 31 day interval position verification.	SR 4.6.1.7.1	SR 3.6.3.1
3.6.3 A.11	Eliminates a cross-reference to a related SR since cross-references are not used in ITS or NUREG-1432 and the LCO for the related SR imposes its own Applicability.	SR 4.6.3.6	LCO 3.6.3
3.6.3 A.12	Rewords a NOTE regarding unisolating penetration flow paths (except for 42 inch purge penetrations) intermittently under administrative control.	LCO 3.6.3 Note *	LCO 3.6.3 NOTE
3.6.3 A.13	Adds clarifying information regarding devices acceptable for isolating the penetration flow path.	LCO 3.6.1.7 Action a, b, c	LCO 3.6.3 Action A, D

Discussion of Change	Description	CTS Section	ITS Section
3.6.3 A.14	Adds a clarifying statement directing user to the appropriate Action for purge valve leakage not within limits.	LCO 3.6.1.7 Action a, b LCO 3.6.3 Action 1	LCO 3.6.3 Action A
3.6.3 A.15	Changes the reference for testing the Containment Spray System automatic valves from CTS Specification 4.0.5 (ASME Section XI pump and valve testing) to the Inservice Testing Program which is described in Section 5.5.8 of ITS and contains the ASME Section XI pump and valve testing requirements.	SR 4.6.3.3	SR 3.6.3.5
3.6.3 A.16	Expands the devices allowed to isolate a penetration by including the use of a check valve with flow through the valve secured which is considered an "automatic valve." "Deactivated automatic valves" were permitted in the CTS source.	LCO 3.6.3 Action 1.b	LCO 3.6.3 Action A.1
3.6.3 A.17	Eliminates the exemption to the provisions of Specification 3.0.4 since the Actions permit operation for an unlimited time and an exemption is not required.	LCO 3.6.3 Action 1.e	LCO 3.0.4
3.6.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.1.4	LCO 3.6.4
3.6.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.1.5	LCO 3.6.5
3.6.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.2.1	LCO 3.6.6
3.6.6 A.2	Eliminates specific reference to restoring operability as a method of exiting the LCO Actions since this is already provided generically in ITS LCO 3.0.2.	LCO 3.6.2.1 Action	LCO 3.0.2
3.6.6 A.3	Adds a new Action that requires immediate entry into LCO 3.0.3 if two containment spray pumps are inoperable. Though not specifically addressed as an Action in CTS, the requirements are the same and therefore this is only an administrative change.	LCO 3.0.3	LCO 3.6.6 Action C

Discussion of Change	Description	CTS Section	ITS Section
3.6.6 A.4	Changes the reference for containment spray pump testing from CTS Specification 4.0.5 (ASME Section XI pump and valve testing) to the Inservice Testing Program which is described in Section 5.5.8 of ITS and contains the ASME Section XI pump and valve testing requirements.	SR 4.6.2.1.b	SR 3.6.6.3
3.6.6 A.5	Moves the information regarding when testing system actuation with test signals from the SR to the Frequency column of ITS but makes no change in intent.	SR 4.6.2.1.d	SR 3.6.6.4
3.6.6 A.6	Rewords the SR for verification of proper valve response upon receipt of a recirculation actuation test signal but does not change the intent of the SR.	SR 4.6.2.1.d.2	SR 3.6.6.4
3.6.6 A.7	Rewords the Action and Completion Time requirements for one containment spray system inoperable and eliminates the restoration of Operability as an explicit action since it is always an available option. The total time allowed and the intent of the Action remain the same.	LCO 3.6.2.1 Action	LCO 3.6.6 Action A, B
3.6.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.4.2	LCO 3.6.7
3.6.7 A.2	Not Used	N/A	N/A
3.6.7 A.3	Not Used	N/A	N/A
3.7.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.1	LCO 3.7.1
3.7.1 A.2	Eliminates the portion of the Action opening statement that simply describes MODE 1 and 2 requirements and adds no useful information.	LCO 3.7.1.1 Action a	LCO 3.7.1 Action A
3.7.1 A.3	Eliminates an unnecessary statement "...operation in MODES 1 and 2 may proceed provided that..." since this is implied by the nature of the Action statement.	LCO 3.7.1.1 Action a	LCO 3.7.1 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.7.1 A.4	Eliminates an unnecessary statement "...either all the inoperable valves are restored to OPERABLE status..." in the Action since this is an method of exiting the Action statement is addressed in LCO 3.0.2 and does not need to be included here.	LCO 3.7.1.1 Action a	LCO 3.7.1 Action A
3.7.1 A.5	Adds a clarifying NOTE which states that "Separate Condition entry is allowed for each MSSV" which was not needed in the CTS source by the nature of its format.	LCO 3.7.1.1 Actions	LCO 3.7.1 Actions
3.7.1 A.6	Makes reference to Table 3.7.1-1 and so eliminates the need to restate the provisions of normal operations in the Actions.	LCO 3.7.1.1 Action b	LCO 3.7.1 Action B
3.7.1 A.7	Eliminates a Specification 3.0.4 exclusion by the addition of a NOTE which states "Not required to be performed prior to entry into MODE 3." The allowance to enter MODE 3 and test MSSVs at hot conditions is retained and the intent is unchanged.	LCO 3.7.1.1 Action c	SR 3.7.1.1
3.7.1 A.8	Eliminates an asterisked note regarding an allowance for Unit 1, Cycle 5 operation at 100% power with one inoperable MSSV.	LCO 3.7.1.1 Table 3.7-2 Note *	LCO 3.7.1
3.7.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.5	LCO 3.7.2
3.7.2 A.2	Changes the reference for stroke time testing of the MSIVs from CTS Specification 4.0.5 (ASME Section XI pump and valve testing) to the Inservice Testing Program which is described in Section 5.5.8 of ITS and contains the ASME Section XI pump and valve testing requirements.	SR 4.7.1.5.1	SR 3.7.2.1
3.7.2 A.3	Eliminates an exclusion to Specification 4.0.4 and adds a NOTE stating that MODE 3 entry is permitted prior to performing the MSIV closure time Surveillance. This permits establishment of conditions under which the acceptance criteria was generated and does not change the intent of the CTS source requirements.	SR 4.7.1.5.2	SR 3.7.2.1

Discussion of Change	Description	CTS Section	ITS Section
3.7.2 A.4	Eliminates an exclusion to Specification 3.0.4 which is unneeded since the Action permits continued operation for an unlimited period of time and the intent is therefore equivalent.	LCO 3.7.1.5 MODE 2, 3, 4 Action. b	LCO 3.7.2 Action C
3.7.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.6.3	LCO 3.7.3
3.7.3 A.2	LCO restated to be equipment specific since the MFIVs are now located in a separate and unique LCO instead of being a subpart of the Containment Isolation Valve LCO.	LCO 3.6.3	LCO 3.7.3
3.7.3 A.3	Adds a clarifying NOTE which states that "Separate Condition entry is allowed for each penetration flow path" which was not needed in the CTS source by the nature of its format.	LCO 3.6.3 Actions	LCO 3.7.3 NOTE
3.7.3 A.4	Retains the requirement to "Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange" since there are no manual valves or blind flanges that can be used for this purpose.	LCO 3.6.3 Action 1.c	LCO 3.7.3
3.7.3 A.5	Eliminates an exclusion to Specification 3.0.4 which is unneeded since the Action permits continued operation for an unlimited period of time and the intent is therefore equivalent.	LCO 3.6.3 Action 1.e	LCO 3.7.3
3.7.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.6	LCO 3.7.4
3.7.4 A.2	Removes cross reference note to a Special Test Exception because cross reference notes are not used in ITS or NUREG-1432.	LCO 3.7.1.6	LCO 3.7.4
3.7.4 A.3	Rewords the LCO to clarify that only one ADV per steam generator is required to be OPERABLE.	LCO 3.7.1.6	LCO 3.7.4
3.7.4 A.4	Rewords the Action Condition from "With less than one atmospheric dump valve..." to "One required ADV line inoperable" but does not change the intent.	LCO 3.7.1.6 Action	LCO 3.7.4 Action A

Discussion of Change	Description	CTS Section	ITS Section
3.7.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.2	LCO 3.7.5
3.7.5 A.2	Changes the LCO and Action statements to refer to "trains" instead of "flowpaths" and "pumps" but does not change the scope or intent.	LCO 3.7.1.2 LCO 3.7.1.2 Action a, b, c	LCO 3.7.5
3.7.5 A.3	Adds clarification that the SR applies to valves the water and both steam flow paths for the AFW pump.	SR 4.7.1.2.a.1	SR 3.7.5.1
3.7.5 A.4	Not Used	N/A	N/A
3.7.5 A.5	Moves the statement directing that testing system actuation with test signals from the Surveillance column of CTS to the Frequency column of ITS but does not change the intent.	SR 4.7.1.2.c	SR 3.7.5.3 SR 3.7.5.4
3.7.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.3	LCO 3.7.6
3.7.6 A.2	Rewords the Action and Completion Time requirements for the condensate storage tank inoperable to eliminate the restoration of Operability as an explicit action since it is always an available option by LCO 3.0.2. The total time allowed and the intent of the Action remain the same.	LCO 3.7.1.3 Action a	LCO 3.7.6 Action A.1
3.7.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.3	LCO 3.7.7
3.7.7 A.2	Provides clarification that the requirement to perform the 18 month EW pump and valve tests does not include the need to perform the testing "during shutdown".	SR 4.7.3.b SR 4.7.3.c	SR 3.7.7.2 SR 3.7.7.3
3.7.7 A.3	Provide clarification that the permissible signals to demonstrate satisfaction of the Surveillance Requirements include both actual and simulated actuation signals.	SR 4.7.3.b SR 4.7.3.c	SR 3.7.7.2 SR 3.7.7.3

Discussion of Change	Description	CTS Section	ITS Section
3.7.7 A.4	Adds a NOTE to Required Action A.1 directing users to the shutdown cooling TS for Actions to take when that system is made inoperable by an inoperable EW System.	LCO 3.7.3	LCO 3.7.7 Action A.1
3.7.7 A.5	Adds a NOTE to SR 3.7.7.1 to clarify that isolation of EW System Flow to individual components does not render EW inoperable.	SR 4.7.7.1	SR 3.7.7.1
3.7.8 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.4	LCO 3.7.8
3.7.8 A.2	Adds a NOTE to SR 3.7.8.1 to clarify that "Isolation of ESPS flow to individual components does not render ESPS inoperable.	SR 4.7.4.1	SR 3.7.8.1
3.7.8 A.3	Adds a NOTE to LCO 3.7.8 Required Action A.1 directing users to the emergency diesel generator TS and shutdown cooling TS for Actions to take when those systems are made inoperable by an inoperable ESPS.	LCO 3.7.4	LCO 3.7.8 Action A.1
3.7.9 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.5	LCO 3.7.9
3.7.9 A.2	Transfers the details of ultimate heat sink OPERABILITY from the LCO to the associated Surveillance Requirements.	LCO 3.7.5	SR 3.7.9.1 SR 3.7.9.2
3.7.9 A.3	Clarifies that the use of the word "average" regarding the verification of the spray pond water temperature is not necessary since there is only a single location for temperature determination and it provides a representative indication of the bulk temperature.	LCO 3.7.5.b	SR 3.7.9.2
3.7.10A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.6	LCO 3.7.10
3.7.10A.2	Provides clarification that the Essential Chilled Water System valve position verification applies to all valves in the flow path not just valves servicing safety-related equipment.	SR 4.7.6.1	SR 3.7.10.1
3.7.11A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.7	LCO 3.7.11

Discussion of Change	Description	CTS Section	ITS Section
3.7.11A.2	Adds a new Action which addresses two CREFS trains inoperable in MODES 1, 2, 3 and 4 and requires immediate entry into LCO 3.0.3. This is equivalent to the CTS approach for this situation and so does not represent a change in intent.	LCO 3.7.7	LCO 3.7.11 Action F
3.7.11A.3	Removes statement of CREFS filter testing and instead makes reference to the Ventilation Filter Test Program which is describe in section 5.5.11 of the ITS.	SR 4.7.7.b	SR 3.7.11.2
3.7.11A.4	Removes the specification how the CREFS System must be started for the monthly test of operation.	SR 4.7.7.a	SR 3.7.11.1
3.7.12A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.7	LCO 3.7.12
3.7.12A.2	Adds a new Action which addresses two CREATCS trains inoperable in MODES 1, 2, 3 and 4 and requires immediate entry into LCO 3.0.3. This is equivalent to the CTS approach for this situation and so does not represent a change in intent.	LCO 3.7.7	LCO 3.7.12 Action F
3.7.13A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.8	LCO 3.7.13
3.7.13A.2	Removes details of testing the ventilation filters and instead makes reference to the Ventilation Filter Test Program which is described in ITS section 5.5.11.	SR 4.7.8.b SR 4.7.8.c SR 4.7.8.d SR 4.7.8.e SR 4.7.8.f	SR 3.7.13.2
3.7.13A.3	Deletes a cross reference to another LCO since cross references are not used in the ITS or NUREG-1432. This is an administrative change only.	LCO 3.7.8 Note *	LCO 3.7.13
3.7.14A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.11	LCO 3.7.14
3.7.14A.2	Deletes a restatement of Applicability in the SR that mirrors the LCO applicability.	SR 4.9.11	SR 3.7.14.1

Discussion of Change	Description	CTS Section	ITS Section
3.7.15A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.13	LCO 3.7.15
3.7.15A.2	Deletes the specification as to how boron concentration of the spent fuel pool is to be determined and clarifies that not only is it to be determined, it must also be within the limits of the specification.	SR 4.9.13	SR 3.7.15.1
3.7.16A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.7.1.4	LCO 3.7.16
3.7.17A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	Section 5.3.1.2	LCO 3.7.17
3.8.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.1	LCO 3.8.1
3.8.1 A.2	Not Used	N/A	N/A
3.8.1 A.3	Eliminates the Action footnotes regarding entry into applicable conditions and requirements because the format change of ITS makes them redundant. See ITS 1.3.	LCO 3.8.1.1 Action c, d, e	LCO 3.8.1 Action C, D, E
3.8.1 A.4	Replaces the term "fuel limited" with "modified" regarding DG starts involving idling and gradual acceleration to synchronous speed.	SR 4.8.1.1.2.a .2 Footnote 2	SR 3.8.1.2 NOTE 3
3.8.1 A.5	Modifies the wording regarding the circuits between the offsite transmission network to the switchyard and the circuits between the switchyard and the diesel generators to improve clarity.	LCO 3.8.1.1.a LCO 3.8.1.1.b	LCO 3.8.1.a LCO 3.8.1.b
3.8.1 A.6	Adds a new Action requiring immediate entry into Specification 3.0.3 if three or more required AC sources are inoperable. This is consistent with the intent of CTS and therefore represents an administrative change only.	LCO 3.8.1.1	LCO 3.8.1 Action I

Discussion of Change	Description	CTS Section	ITS Section
3.8.1 A.7	Adds a statement for two SRs that states "All DG starts may be preceded by an engine prelube period." which is consistent with the BASES and therefore an administrative change only.	SR 4.8.1.1.2.d .8 SR 4.8.1.1.2.e	SR 3.8.1.15 SR 3.8.1.20
3.8.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.2	LCO 3.8.2
3.8.2 A.2	Clarifies the LCO wording regarding the circuits between the offsite transmission network, the onsite Class 1E distribution system and diesel generator. Also ensures all required loads are powered from offsite power and that a diverse power source is available if the offsite circuit is lost.	LCO 3.8.1.2.a LCO 3.8.1.2.b	LCO 3.8.2.a LCO 3.8.2.b
3.8.2 A.3	Transfers three SRs (4.8.1.1.2.d.4, 4.8.1.1.2.d.5 and 4.8.1.1.2.d.10) from the note pertaining to SRs that are not required and includes them in the list of ITS SRs that are not applicable because ESF functions are not require OPERABLE during shutdown.	SR 4.8.1.2 Note 2	SR 3.8.2.1 NOTE
3.8.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.1.3.1	LCO 3.8.3
3.8.3 A.2	Adds a new Action to declare the associated DG inoperable "immediately" if the Required Action and Associated Completion Times of the other Actions are not met or if another reason for inoperability is identified that is not addressed in the other LCO Actions.	LCO 3.8.1.3.1	LCO 3.8.3 Action F
3.8.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.2.1	LCO 3.8.4
3.8.4 A.2	Removes the use of the word "energized" in regard to the LCO requirement that the DC trains be energized but does not change the intent.	LCO 3.8.2.1	LCO 3.8.4

Discussion of Change	Description	CTS Section	ITS Section
3.8.4 A.3	Removes the use of the word "Required" in connection with DC trains to eliminate confusion but does not change the intent.	LCO 3.8.2.1 Action a	LCO 3.8.4 Action A
3.8.4 A.4	Moves the direction to perform the Surveillance during times when the plant is not operating by deleting "during shutdown" and adding a Surveillance NOTE directing that the SR not be performed in MODES 1 through 4. The intent is not changed.	SR 4.8.2.1.d SR 4.8.2.1.e	SR 3.8.4.7 SR 3.8.4.8
3.8.4 A.5	Removes the use of the phrase "actual and simulated" regarding loads in verification of the battery capacity but does not change the intent.	SR 4.8.2.1.d	SR 3.8.4.7
3.8.4 A.6	Not Used	N/A	N/A
3.8.4 A.7	Changes the phrase "DC trains inoperable" to "DC electrical power subsystems (exclusive of the battery charge) inoperable" to more clearly define the meaning, eliminate confusion and make a clear distinction between the battery chargers and the rest of the DC power subsystem.	LCO 3.8.2.1	LCO 3.8.4 Action A
3.8.4 A.8	Not Used	N/A	N/A
3.8.4 A.9	Adds the phrase "that would cause performance degradation" to the SR verification requirement that no visual indication of physical damage or abnormal deterioration exists.	SR 4.8.2.1.c.1	SR 3.8.4.3
3.8.4 A.10	Deletes specification of the method in which restoring charging capacity must be done.	LCO 3.8.2.1 Action (b)	LCO 3.8.4 Action C
3.8.4 A.11	CTS specifies criteria for both Exide and AT&T brand batteries. In PVNGS letter 102-04053 dated December 17, 1997, PVNGS requested to change the term Exide to reference the more descriptive term "low specific gravity cells." This is characterized as an administrative change since the battery performance criteria and cell parameters are not changed from those specified in the CTS. (See draft SE Section III.G on removing Exide and AT&T from CTS.)	SR 4.8.2.1	SR 3.8.4.1 SR 3.8.4.8
3.8.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.2.2	LCO 3.8.5

Discussion of Change	Description	CTS Section	ITS Section
3.8.5 A.2	Removes the use of the word "energized" in connection with the LCO describing the operability requirements for the DC train. The bases description details that the buses must be connected to batteries and operating chargers.	LCO 3.8.2.2	LCO 3.8.5
3.8.5 A.3	Deletes specification of the method in which restoring charging capacity must be done.	LCO 3.8.2.2 Action (b)	LCO 3.8.5 Action B
3.8.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.2.1	LCO 3.8.6
3.8.6 A.2	Removes references to Exide batteries which are no longer used at PVNGS. (See draft SE Section III.G on removing Exide and AT&T from CTS.)	LCO 3.8.2.1 Table 4.8-2	LCO 3.8.6 Table 3.8.6-1
3.8.6 A.3	Changes the reference to the number of connected cells to be tested from each battery back from 6 to 10% (which is 6 cells since the bank consists of 60 cells). This is an administrative change only.	SR 4.8.2.1.b.3	SR 3.8.6.3
3.8.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.3.1	LCO 3.8.7
3.8.7 A.2	Reflects the effect of the administrative split of the source (CTS) Onsite Power Distribution Systems - Operation into two ITS Specifications, 3.8.7 Inverters - Operating and 3.8.9 Distributing Systems - Operating System. Adjustments were made in phrasing to accommodate this split but the intent is unchanged.	LCO 3.8.3.1 Action b.2	LCO 3.8.7 Action A.1, B.1, B.2
		LCO 3.8.3.1 Action b.1	LCO 3.8.9 Action B.1, D.1, D.2
3.8.8 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.3.2	LCO 3.8.8

Discussion of Change	Description	CTS Section	ITS Section
3.8.8 A.2	Reflects the effect of the administrative split of the source (CTS) Onsite Power Distribution System - Shutdown into two ITS Specifications, 3.8.8 Inverters - Shutdown and 3.8.10 Distribution Systems - Shutdown. Adjustments were made in phrasing to accommodate this split but the intent is unchanged.	LCO 3.8.3.2.b (Inverters)	LCO 3.8.8
3.8.8 A.2	Reflects the effect of the administrative split of the source (CTS) Onsite Power Distribution System - Shutdown into two ITS Specifications, 3.8.8 Inverters - Shutdown and 3.8.10 Distribution Systems - Shutdown. Adjustments were made in phrasing to accommodate this split but the intent is unchanged.	LCO 3.8.3.2.a LCO 3.8.3.2.b LCO 3.8.3.2.c	LCO 3.8.10
3.8.9 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.3.1	LCO 3.8.9
3.8.9 A.2	Explicitly states that the AC vital bus can be powered from the inverter or the Class 1E constant voltage regulator.	LCO 3.8.3.1 LCO 3.8.3.1 Action c	LCO 3.8.9 Bases
3.8.10A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.8.3.2	LCO 3.8.10
3.8.10A.2	Explicitly states that the AC vital bus can be powered from the inverter or the Class 1E constant voltage regulator.	LCO 3.8.3.2.b	LCO 3.8.10 Bases LCO 3.8.9 Bases
3.9.1 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.1	LCO 3.9.1
3.9.1 A.2	Deletes the redundant statement in the MODE 6 Applicability regarding reactor vessel head bolts not fully tight or the head removed which are unnecessary since the ITS Definition of MODE 6 already addresses this.	LCO 3.9.1	LCO 3.9.1

Discussion of Change	Description	CTS Section	ITS Section
3.9.1 A.3	Not Used	N/A	N/A
3.9.1 A.4	Provides clarification that the specific requirement for the boron concentration of all filled portions of the Reactor Coolant System to be maintained "uniform" is part of the normal activities related to shutdown cooling and refueling.	LCO 3.9.1	LCO 3.9.1
3.9.1 A.5	Deletes the specific requirement that "chemical analysis" be used to determine boron concentration of the Reactor Coolant System and adds the requirement that the concentration be determined to be within limits. This was the intent of the original CTS specification so the change is administrative only.	SR 4.9.1.2	SR 3.9.1.1
3.9.2 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.2	LCO 3.9.2
3.9.2 A.2	Adds a NOTE which excludes neutron detectors from the requirement for CHANNEL CALIBRATION.	SR 4.9.2.b SR 4.9.2.c	SR 3.9.2.2
3.9.3 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.4 LCO 3.9.9	LCO 3.9.3
3.9.3 A.2	Deletes a redundant specification of the status required for containment building penetrations.	SR 4.9.4	SR 3.9.3.1
3.9.3 A.3	Eliminates a cross reference which is no longer needed since the referenced material (SR 4.9.9) has been combined with the referencing SR into this ITS LCO.	SR 4.9.4.b	LCO 3.9.3
3.9.3 A.4	Clarifies the nomenclature for devices that may be used to close piping penetrations (other than purge valve penetrations) which provide direct access from the containment atmosphere to outside atmosphere.	LCO 3.9.4.c.1	LCO 3.9.3.c.1
3.9.3 A.5	Combines the requirements of two different CTS LCOs addressing containment purge penetration closure and the containment purge and exhaust isolation system to require that containment purge penetrations be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. The intent is not changed.	LCO 3.9.4.c.2 LCO 3.9.9	LCO 3.9.3.c.1

Discussion of Change	Description	CTS Section	ITS Section
3.9.4 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.8.1	LCO 3.9.4
3.9.4 A.2	Rewords the Action regarding the reload of irradiated fuel from "suspend all operations involving an increase in the reactor decay heat load..." to suspending loading of irradiated fuel assemblies in the core. Also changes the logical connector from "or" to "and" for clarification but does not alter the intent.	LCO 3.9.8.1 Action	LCO 3.9.4 Action A.1, A.2
3.9.5 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.8.2	LCO 3.9.5
3.9.5 A.2	Rewords the Action regarding the number of shutdown cooling loops OPERABLE from "With less than the required..." to "One SDC loop..." but since there are two and both are required, the meaning is equivalent and the intent is not changed.	LCO 3.9.8.2 Action a	LCO 3.9.5 Action A
3.9.5 A.3	Rewords the Action regarding shutdown cooling from "With no shutdown cooling loop in operation..." to "No SDC loop OPERABLE or in operation." The LCO requires two loops to be OPERABLE, one of which must be in operation. The Actions are equivalent and the rewording is for clarity only.	LCO 3.9.8.2 Action b	LCO 3.9.5 Action B
3.9.5 A.4	Eliminates the Required Action for suspending operations involving an increase in reactor decay heat load (irradiated fuel assembly movement). This is not needed since movement of those assemblies is not permitted when this LCO is Applicable. The Applicability of this LCO is MODE 6 with water level < 23 feet above the top of the flange. The Refueling Water Level - Fuel Assemblies LCO requires the level to be at or above 23 feet before fuel assembly movement is permitted.	LCO 3.9.8.2 Action b	LCO 3.9.5
3.9.5 A.5	Rewords the Action to clarify that the intention is to immediately suspend operations involving a reduction in reactor coolant boron concentration when no SDC loop is in operation.	LCO 3.9.8.2 Action b	LCO 3.9.5 Action B.1

Discussion of Change	Description	CTS Section	LTS Section
3.9.6 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.10.1	LCO 3.9.6
3.9.6 A.2	Rewords the Action from "...suspend all operations involving movement of fuel assemblies..." to "Suspend movement of fuel assemblies within containment" for clarification but does not change the intent.	LCO 3.9.10 Action	LCO 3.9.6 Action A.1
3.9.7 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 3.9.10.2	LCO 3.9.7
3.9.7 A.2	Rewords the LCO to clarify that the water level restriction is applicable only if there is irradiated fuel in the reactor vessel but does not change the intent.	LCO 3.9.10.2	LCO 3.9.7
4.0 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	Section 5.0	Section 4.0
4.0 A.2	Moves the descriptive wording pertaining to spent fuel pool loading patterns.	Section 5.3.1.2 (a through c)	Section 4.3.1.1. (d through f)

Discussion of Change	Description	CTS Section	ITS Section
5.0 A.1	Reformatted and renumbered in accordance with NUREG-1432, Rev. 1 to improve readability. Changes included editorial rewording and the insertion of additional descriptive information.	LCO 4.0.5 LCO 3.4.4 LCO 3.4.8.3 LCO 3.5.2 LCO 3.6.1.3 LCO 3.6.1.6 LCO 3.6.3 LCO 3.6.4.3 LCO 3.7.7 LCO 3.7.8 LCO 3.11.1 LCO 3.11.2 LCO 3.11.3 Section 6.0	Section 5.0
5.0 A.2	Eliminates the requirement that the Vice President-Nuclear Production issue an annual directive stating that the Shift Supervisor is responsible for the Control Room function. This is an unnecessary reminder and the responsibilities of the Shift Supervisor are adequately documented elsewhere in the ITS and UFSAR.	Section 6.1.2	Section 5.1.2
5.0 A.3	Eliminates specification that organizational charts in the UFSAR be updated in accordance with 10 CFR 50.71(e) since that document includes the requirements for updating the UFSAR and restatement in Technical Specifications is redundant.	Section 6.2.1.a	Section 5.2.1.a
5.0 A.4	Rewords the description of responsibilities for the Department Leader Operations in regard to who directs the licensed activities of the licensed operators, clearly establishing that position as responsible for overall unit operation.	Section 6.3.1	Section 5.1.1
5.0 A.5	Rewords the statement regarding who is authorized to approve proposed modifications to nuclear-safety related structures, systems and components. CTS placed it with the Department Leader, Operations or Director, Site Operations. ITS places it with the Department Leader, Operations (who works for the Director, Site Operations) or his designee.	Section 6.5.2.3	Section 5.1.1

Discussion of Change	Description	CTS Section	ITS Section
5.0 A.6	Eliminates referral to specific procedures that must written because they are elsewhere required by ITS 5.4.1.a through referral to Regulatory Guide 1.33.	Section 6.8.1 (b, c, h, k)	Section 5.4.1.a
5.0 A.7	Eliminates redundant sections requiring written procedures for ODCM implementation, secondary water chemistry program implementation and PASS implementation. Section 5.4.1.e of ITS requires written procedures for all programs addressed in ITS section 5.5 and these named items are addressed in that location.	Section 6.8.1 (i, l and m)	Section 5.4.1.e
5.0 A.8	Updates the section references of 10 CFR 20 to reflect the latest version (20.1302, 20.1601 and table 2).	Section 6.4.g (2) and (3)	Section 5.5.4 (b and c)
5.0 A.9	Rewords the reference to 10 CFR 50.4 regarding report submittal but does not change the intent.	Section 6.9.1	Section 5.6
5.0 A.10	Changes the submittal dates for the Annual Radiological Environmental Operating Report and Occupational Radiation Exposure Report.	Sections 6.9.1.4 6.9.1.7	Sections 5.6.1 5.6.2
5.0 A.11	Eliminates an unnecessary referral to the submittal site for various reports. The ITS requires submittal in accordance with 10 CFR 50.4 which provides the NRC distribution requirements for report submittal.	Sections 6.9.1.6 6.9.1.10	Sections 5.6.4 5.6.5.b
5.0 A.12	Deletes an unnecessary requirement to submit special reports to "the NRC within the time period specified for each report." Each special report contains requirements for submittal.	Section 6.9.2	Section 5.6
5.0 A.13	Deletes the "± 0.5" from the test pressure requirement for testing air locks at ≥14.5 ± 0.5 psig which is a redundant requirement included in the acceptance criteria.	Section 6.16.b.2	Section 5.5.16
5.0 A.14	Clarifies that ASME Boiler and PV Code Section XI testing includes "applicable supports."	LCO 4.0.5	Section 5.5.8
5.0 A.15	Deletes an unnecessary statement that says Inservice Inspection Program and Testing activities must be performed in addition to other Surveillance requirements.	LCO 4.0.5.d	Section 5.5.8

Discussion of Change	Description	CTS Section	ITS Section
5.0 A.16	Adds a statement that SR 3.0.3 is applicable to ITS 5.5.8 for clarification since the ITS Applicability SRs are not normally applied to frequencies identified in Administrative Section of the ITS.	LCO 4.0.5	Section 5.5.8.c
5.0 A.17	Adds an ASME biennial (2 year) testing interval with a 731 day frequency. This frequency is already identified in the ASME code and this change is therefore administrative.	LCO 4.0.5.b	Section 5.5.8.a
5.0 A.18	Rewords the requirement to submit a special report to the Commission to document challenges to the SCS suction line relief valves or pressurizer safety valves.	LCO 3.4.8.3 Action e	Section 5.6.4
5.0 A.19	Adds a statement that SR 3.0.2 and SR 3.0.3 are applicable to ITS 5.5.6 for clarification since the ITS Applicability SRs are not normally applied to frequencies identified in Administrative Section of the ITS.	SR 4.6.1.6.1	Section 5.5.6
5.0 A.20	Transfers the requirements for the Liquid Holdup Tanks, Explosive Gas Mixture and Gas Storage Tanks to ITS Section 5.5.12 to be housed within the ITS Explosive Gas and Storage Tank Radioactivity Monitoring Program.	LCO 3.11.1 LCO 3.11.2 LCO 3.11.3	Section 5.5.12
5.0 A.21	Expands on the definition of the word "temporary" regarding what precisely constitutes a temporary radwaste tank.	LCO 3.11.1	Section 5.5.12.c
5.0 A.22	Adds a statement that SR 3.0.2 and SR 3.0.3 is applicable to ITS 5.5.12 for clarification since the ITS Applicability SRs are not normally applied to frequencies identified in Administrative Section of the ITS.	LCO 3.11.1 Action b LCO 3.11.2 Action c LCO 3.11.3 Action b	Section 5.5.12
5.0 A.23	Adds a statement that SR 3.0.2 and SR 3.0.3 is applicable to ITS 5.5.11 for clarification since the ITS Applicability SRs are not normally applied to frequencies identified in Administrative Section of the ITS.	LCO 3.7.8	Section 5.5.11

Discussion of Change	Description	CTS Section	ITS Section
5.0 A.24	Rewords the sample analysis requirement for diesel fuel storage tank samples from checking "viscosity and sediment" to "particulate concentration" which are equivalent. Both check for fuel oil degradation and the change is therefore administrative.	SR 4.8.1.3.1.2	Section 5.5.13.c
5.0 A.25	Changes the calculated containment peak pressure for the design basis loss of coolant accident (P_a) from 49.5 psig to 52 psig. This change is characterized as administrative because the change reflects correct design pressure and was approved by TS Amendments 113, 106 and 85 to Units 1, 2, and 3, respectively.	LCO 3.6.1.3.b Section 6.16	Section 5.5.16
5.0 A.26	Adds clarification in the form of the statement "Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m)."	Section 6.2.2.a Table 6.2-1	Section 5.2.2.a
	Adds the clarifying statement "For the purpose of 10 CFR 50.54, a licensed senior reactor operator (SRO) and a licensed reactor operator (RO) are individuals who, in addition to meeting the requirements of 5.3.1, perform the functions described in 10 CFR 50.54(m).	Section 6.3.1	Section 5.3.2
5.0 A.27	Revises content to incorporate changes to 10 CFR 20 and 10 CFR 50.36a intended to eliminate possible problems with implementation of the revised 10 CFR 20 requirements. Revised to ensure consistency with regulations.	Section 6.8.4.g (2), (7), (10)	Section 5.5.4.6

Table of PVNGS More Restrictive Changes (M)

Discussion of Change	Summary of Change	CTS Section	ITS Section
1.0 M.1	Not Used	N/A	N/A
1.0 M.2	Requires the required instrument display to be tested as part of the CHANNEL CALIBRATION and for in-place cross-calibration whenever an RTD is replaced.	1.4 Definition: CHANNEL CALIBRATION	1.1 Definition: CHANNEL CALIBRATION
		1.6.a Definition: CHANNEL FUNCTIONAL TEST	1.1 Definition: CHANNEL FUNCTIONAL TEST
1.0 M.3	Not Used	N/A	N/A
1.0 M.4	Increased requirement related to the Completion Times for situations when two subsystems become inoperable concurrently (without a note that allows the Conditions to be entered separately).	1.0 General	1.3 DESCRIPTION
2.0 None	N/A	N/A	N/A
3.0 M.1	Clarification that the interval extension of 1.25 times the specified interval does not apply to Frequencies specified as "once."	SR 4.0.2	SR 3.0.2
3.0 M.2	Requirement that missed Surveillance with an LCO Required Action Completion time less than 24 hours and Frequency less than 24 hours be performed within the more restrictive time of either 24 hours or the limit of the specified frequency.	SR 4.0.3	SR 3.0.3
3.1.1 None	N/A	N/A	N/A
3.1.2 M.1	Added Action for the condition where reactor criticality could be achieved by shutdown group CEA movement.	LCO 3.1.1.2.c	LCO 3.1.2 Action B
3.1.3 None	N/A	N/A	N/A
3.1.4 None	N/A	N/A	N/A
3.1.5 M.1	Requirement that with two or more CEAs misaligned by > 9.9 inches the reactor trip breakers be opened.	LCO 3.1.3.1 Action b	LCO 3.1.5 Action E
3.1.6 None	N/A	N/A	N/A
3.1.7 M.1	Requirement that the PDIL alarm circuit be OPERABLE.	SR 4.1.3.6	LCO 3.1.7

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.1.7 M.1	Action requirement for inoperable PDIL alarm circuit.	SR 4.1.3.6	LCO 3.1.7 Action D
	Requirement to verify PDIL alarm circuit OPERABILITY.	SR 4.1.3.6	SR 3.1.7.3
3.1.8 None	N/A	N/A	N/A
3.1.9 M.1	MODE applicability changed to MODES 2 and 3 during PHYSICS TESTS and no longer includes MODE 4.	LCO 3.10.1	LCO 3.1.9 APPLICABILITY
3.1.10 M.1	Required Action is added to reduce power to less than or equal to the test power plateau within 15 minutes if power exceeds the test thermal plateau.	LCO 3.10.2	LCO 3.1.10 Action A
3.1.10 M.2	Not Used	N/A	N/A
3.1.11 M.1	Elimination of CEA Position and Shutdown CEA Insertion Limit specification suspension since it is not necessary.	LCO 3.10.4	LCO 3.1.11
3.1.11 M.1 (continued)	Elimination of CEA Position and Shutdown CEA Insertion Limit specification suspension since it is not necessary.	SR 4.10.4.2	LCO 3.1.11
3.1.11 M.2	Change in MODE applicability limits for LCO reducing circumstances when ITS 3.1.11 can be invoked.	LCO 3.10.4	LCO 3.1.11 APPLICABILITY
3.1.11 M.3	Addition of requirements of specification 3.2.4, DNBR.	LCO 3.10.4	LCO 3.1.11
		SR 4.10.4.2	SR 3.1.11.1
3.2.1 None	N/A	N/A	N/A
3.2.2 None	N/A	N/A	N/A
3.2.3 M.1	Required Action to reduce the Variable Overpower Trip Setpoint to $\leq$ 55% Rated Thermal Power is added to the ITS.	LCO 3.2.3 Action b.2	LCO 3.2.3 Action B.2
3.2.4 None	N/A	N/A	N/A
3.2.5 None	N/A	N/A	N/A
3.3.1 M.1	Removes NOTE allowing the Local Power Density - High, Departure from Nucleate Boiling Ratio - Low and Logarithmic Power Level - High to be bypassed pursuant to Special Test Exception 3.10.3.	Table 3.3-1 Note (d)	N/A

Discussion of Change	Summary of Change	CTS Section	ITS Section
	Removes NOTE allowing the setpoint for Reactor Coolant Flow - Low (Rate, Floor and Band) to be altered to disable the trip function during testing pursuant to Special Test Exception 3.10.3.	Table 2.2-1 Note (7)	N/A
3.3.1 M.2	Requires adjustment of the linear power level to agree with the calorimetric calculation if the absolute difference is equal to 2% (or greater).	Table 4.3-1 Note (2)	SR 3.3.1.4
3.3.1 M.3	Requires testing within 12 hours after reaching required thermal power level.	Table 4.3-1 Note (3), (7), (8)	SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.6
3.3.2 M.1	Removal of option to make the CPCs OPERABLE in MODES 3*, 4* and 5* and requires that the Logarithmic Power Level - High Setpoints be reduced.	Table 3.3-1 1.B.2.a	Table 3.3.2-1 1.
		Table 3.3-1 1.C.2	Table 3.3.2-1 1.
		Table 3.3-1 Action 10	Table 3.3.2-1 1.
3.3.2 M.2	Requires opening of all RTCBs within 1 hour placing the Unit in MODE 3.	Table 3.3-1 Actions	LCO 3.3.2 Action E
3.3.3 None	N/A	N/A	N/A
3.3.4 M.1	Requires that all six channels of The RPS Matrix Logic must be OPERABLE. Includes a note regarding the applicability of the Action if three RPS Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies.	CTS Table 3.3-1 Item II.A	LCO 3.3.4 LCO 3.3.4 Action A
3.3.5 None	N/A	N/A	N/A
3.3.6 M.1	Requires that all six channels of the ESFAS Matrix Logic must be OPERABLE.	Table 3.3 Item I.B	LCO 3.3.6 LCO 3.3.6 Action A
		Table 3.3 Item III.B.1	LCO 3.3.6 LCO 3.3.6 Action A

Discussion of Change	Summary of Change	CTS Section	ITS Section
		Table 3.3 Item IV.B	LCO 3.3.6 LCO 3.3.6 Action A
		Table 3.3 Item V.B	LCO 3.3.6 LCO 3.3.6 Action A
3.3.6 M.1	Requires that all six channels of the ESFAS Matrix Logic must be OPERABLE.	Table 3.3 Item VI.B	LCO 3.3.6 LCO 3.3.6 Action A
		Table 3.3 Item VII.B	LCO 3.3.6 LCO 3.3.6 Action A
3.3.6 M.2	Requires reducing plant status to MODE 5 for inoperable SIAS, CIAS, and RAS Function Automatic Actuation Logic.	Table 3.3-3 Action 16	LCO 3.3.6 Action F
3.3.7 M.1	Requires Diesel Generator Loss of Voltage (LOV) Start to be OPERABLE in MODE 4 and when required by ITS LCO 3.8.2	Table 3.3-3 Item VIII	LCO 3.3.7 Applicability
	Requires Diesel Generator Loss of Voltage (LOV) Start OPERABLE when required by ITS LCO 3.8.2	Table 3.3-4 Item VIII	LCO 3.3.7 Applicability
		Table 4.3-2 Item VIII	LCO 3.3.7 Applicability
3.3.7 M.2	Requires performance of Delay Time testing for both channels of Loss of Voltage Relays and Degraded Voltage Relays at 18 month intervals and specification of values.	SR 4.3.2.3	SR 3.3.7.3
		Table 3.3-4 Item VIII.A	SR 3.3.7.3
		Table 3.3-4 Item VIII.B	SR 3.3.7.3

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.3.7 M.3	Requires that the applicable LCO for the associated DG (made INOPERABLE) be entered if unable to comply with Required Action B.2.	LCO 3.3.2	LCO 3.3.7 Action B
3.3.7 M.4	Requires that both the Degraded Voltage and Under Voltage Relays be restored to operability within one hour.	LCO 3.3.2 Action 19.a	LCO 3.3.7 Action C
3.3.8 M.1	Not Used	N/A	N/A
3.3.8 M.2	Requires entry into ITS LCO 3.6.3 CONTAINMENT ISOLATION VALVE if the required Action and associated Completion Time of ITS 3.3.8 Condition A is not met.	LCO 3.3.3.1 Table 3.3-6 Item 1.D	LCO 3.3.8 Action B
3.3.8 M.3	Requires a CHANNEL FUNCTIONAL TEST of the CPIAS actuation logic channel	SR 4.4.9	SR 3.3.8.3
	Requires a CHANNEL FUNCTIONAL TEST of the CPIAS manual trip.	SR 4.4.9	SR 3.3.8.5
3.3.8 M.4	Deletes allowance that radiation monitoring channel alarm/trip setpoints can be outside of the allowable values for up to 4 hours.	LCO 3.3.3.1 Action a	LCO 3.3.8
3.3.8 M.5	Deletes exemptions for Specifications 3.0.3 and 3.0.4.	LCO 3.3.3.1 Action c	LCO 3.3.8
3.3.8 M.6	Specifies requirement for CPIAS Manual Trip and Actuation Logic in MODES 1, 2, 3 and 4.	LCO 3.3.3.1	LCO 3.3.8
3.3.9 M.1	Deletes allowance that radiation monitoring channel alarm/trip setpoints can be outside of the allowable values for up to 4 hours.	LCO 3.3.3.1 Action a	LCO 3.3.9
3.3.9 M.2	Deletes exemptions for Specifications 3.0.3 and 3.0.4.	LCO 3.3.3.1 Action c	LCO 3.3.9
		LCO 3.3.2 Table 3.3-3 Notation.*	LCO 3.3.9
3.3.9 M.3	Requires a CHANNEL FUNCTIONAL TEST of the CREFAS actuation logic channel.	LCO 3.3.3.1	SR 3.3.9.3

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.3.9 M.4	Requires a CHANNEL FUNCTIONAL TEST of the CREFAS manual trip channel.	LCO 3.3.3.1	SR 3.3.9.5
3.3.9 M.5	Requires that CREFAS be OPERABLE during the movement of irradiated fuel assemblies.	LCO 3.3.3.1 Table 3.3-6 Item 2.B	LCO 3.3.9
		LCO 3.3.3.1 Table 4.3-3 Item 2.B	LCO 3.3.9
3.3.9 M.6	Adds additional Actions for inoperable CREFAS equipment in MODES 5, 6 and during the movement of irradiated fuel assemblies.	LCO 3.3.3.1 Action 26	LCO 3.3.9 Action C
		LCO 3.3.2 Action 18	LCO 3.3.9 Action C
3.3.9 M.7	Requires that essential ventilation be placed in operation within one hour.	LCO 3.3.2 Action 18	LCO 3.3.9 Action A
3.3.10M.1	Requires two channels of CETs per quadrant, each channel consisting of two sensors.	LCO 3.3.3.6 Table 3.3-10 Item 14	LCO 3.3.10 Table 3.3.10-1 Items 14, 15, 16 and 17
3.3.10M.2	Adds three additional functions (Containment Isolation Valve Position Indication, Reactor Coolant System Activity Indication and Condensate Storage Tank Level Indication) to the list of Post Accident Monitoring (PAM) Instrumentation.	LCO 3.3.3.6	LCO 3.3.10 Table 3.3.10-1 Items 8, 13 and 20
3.3.10M.3	Moves the Containment Hydrogen Monitors to the PAM specification and adds an additional MODE requirements (MODE 3)	LCO 3.6.4.1	LCO 3.3.10 Table 3.3.10-1 Item 10
3.3.10M.4	Requires the plant to be in MODE 4 within 12 hours with both hydrogen monitors inoperable.	LCO 3.6.4.1 Action b	LCO 3.3.10 Table 3.3.10-1 Item 10 (Action F)
3.3.11M.1	Requires OPERABILITY of the Remote Shutdown System in MODE 3.	LCO 3.3.3.5	LCO 3.3.11

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.3.11M.2	Requires the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours.	LCO 3.3.3.5 Action a	LCO 3.3.11
		LCO 3.3.3.5 Action b	LCO 3.3.11
3.3.12M.1	Does not exclude Specification 3.0.3.	LCO 3.1.2.7 Action c	LCO 3.3.12
3.3.12M.2	Adds the requirement for a CHANNEL CALIBRATION for the BDAS.	SR 4.1.2.7	SR 3.3.12.3
3.3.12M.3	Adds action to require suspension of all operations involving positive reactivity additions if RCS Boron Concentration can not be monitored periodically.	LCO 3.1.2.6 Action a.1	LCO 3.3.12 Action B
3.4.1 M.1	Requires MODE 1 performance of surveillance measuring RCS flow rate under normal operating conditions at power with all RCPs running.	SR 4.2.5	SR 3.4.1.3
3.4.2 None	N/A	N/A	N/A
3.4.3 M.1	Excludes CTS 30 minute allowance to restore temperature and/or pressure to within limits following their violation, and 36 hour allowance to reduce pressure if 30 minute requirement not met.	LCO 3.4.8.1 Action	LCO 3.4.3 Action C
3.4.4 None	N/A	N/A	N/A
3.4.5 M.1	Requires immediate suspension of all operations involving a reduction of boron concentration when no RCS loop is in operation.	LCO 3.4.1.2 Action b	LCO 3.4.5 Action C.1
3.4.5 M.2	Restricts the amount of time that all RCPs may be de-energized to one hour per eight hour period.	LCO 3.4.1.2 Footnote *	LCO 3.4.5 Note
3.4.5 M.3	Imposes Action requirements when no reactor coolant loops are OPERABLE instead of just when no reactor coolant loop is in operation.	LCO 3.4.1.2 Action b	LCO 3.4.5 Action C
3.4.6 M.1	Requires immediate suspension of all operations involving a reduction of boron concentration when no RCS loop is in operation.	LCO 3.4.1.3 Action b	LCO 3.4.6 Action C.1
3.4.6 M.2	Restricts the amount of time that all RCPs may be de-energized to one hour per eight hour period.	LCO 3.4.1.3 Footnote *	LCO 3.4.6 Note 1

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.4.6 M.3	Requires verification of correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	SR 4.4.1.3.1	SR 3.4.6.3
3.4.7 M.1	Requires verification of correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	LCO 3.4.1.4.1	SR 3.4.7.3
3.4.7 M.2	Restricts the amount of time that all RCPs may be de-energized to one hour per eight hour period.	LCO 3.4.1.4.1	LCO 3.4.7 Note 1
3.4.7 M.3	Requires immediate suspension of all operations involving a reduction of boron concentration when no RCS loop is in operation.	LCO 3.4.1.4.1 Action b	LCO 3.4.7 Action B.1
3.4.8 M.1	Introduces an additional requirement to the LCO Note that allows the SDC loops to be de-energized that disallows all draining operations that would further reduce the RCS water volume.	LCO 3.4.1.4.2 Note *	LCO 3.4.8 Note 1
3.4.8 M.2	Requires verification of correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	LCO 3.4.1.4.2	SR 3.4.8.2
3.4.8 M.3	Requires immediate suspension of all operations involving a reduction of boron concentration when no RCS loop is in operation.	LCO 3.4.1.4.2 Action b	LCO 3.4.8 Action B.1
3.4.8 M.4	Restricts the amount of time that all RCPs may be de-energized to one hour per eight hour period.	LCO 3.4.1.4.2 Note *	LCO 3.4.8 Note 1
3.4.8 M.5	Requires immediate suspension of all operations involving a reduction of boron concentration when no RCS loop is in operation.	LCO 3.4.1.4.2 Action b	LCO 3.4.8 Action B
3.4.9 M.1	Removes one hour allowance to restore pressurizer level before entering 6 hour period to transition to MODE 3	LCO 3.4.3.1 Action b	LCO 3.4.9 Action A
3.4.10M.1	Additional requirement that following testing the lift settings for the pressurizer PSV's be within +/- 1% of the specified value.	4.4.2.2	SR 3.4.10.1

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.4.11M.1	Additional SR to ensure that the required Shutdown Cooling System suction line relief valve is OPERABLE.	LCO 3.4.2.1	LCO SR 3.4.11.2
		LCO 3.4.2.1	LCO SR 3.4.11.3
3.4.11M.2	Satisfaction of the Action statement requires the plant to be not only placed in MODE 4, but the additional requirement of any RCS cold leg temperatures ≤ being 214°F. during cooldown or ≤ 291°F. during heatup.	LCO 3.4.2.1	LCO 3.4.11 Action A.3
3.4.11M.3	Additional requirement that following testing the lift settings for the pressurizer PSV's be within +/- 1% of the specified value.	4.4.2.1	SR 3.4.11.1
3.4.12M.1	Changes the end-state MODE (less than 385 psia) for Pressurizer Vents required Action to place the plant in a condition where the LCO no longer applies.	LCO 3.4.10	LCO 3.4.12 Action C
3.4.13None	N/A	N/A	N/A
3.4.14None	N/A	N/A	N/A
3.4.15M.1	Allows the use of the second in-line PIV to fulfill isolation requirements, but requires that the RCS PIV be restored to within limits within 72 hours.	LCO 3.4.5.2	LCO 3.4.15 Action A
3.4.15M.2	Requires entry into applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.	LCO 3.4.5.2	LCO 3.4.15 Actions Note 2
3.4.15M.3	Requires that each valve used to satisfy Required Action A.1 and Required Action A.2 be verified to meet SR 3.4.15.1 and be on the RCS pressure boundary.	LCO 3.4.5.2	LCO 3.4.15 Action A
3.4.16M.1	Requires that SR 3.4.14.1 (RCS water inventory balance) be performed once per 24 hours when the required containment sump monitor or containment atmosphere radioactivity monitor is inoperable.	LCO 3.4.5.1	LCO 3.4.16 Action A
3.4.16M.2	Requires that LCO 3.0.3 entry be made when all required monitors are inoperable.	LCO 3.4.5.1	LCO 3.4.16 Action D

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.4.16M.3	Requires that both the gaseous and particulate containment atmospheric radiation monitors be OPERABLE.	3.4.5.1a	3.4.16.b
3.4.17M.1	Requires the performance of the surveillance to verify RCS DOSE EQUIVALENT I-131 specific activity ≤ 1.0 microcurie/gram in addition to unit SHUTDOWN to MODE 3 with T_{cold} < 500°F.	LCO 3.4.7 Action B	LCO 3.4.17 Action C.1
3.4.17M.2	Requires E-Bar surveillance performance within 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours should the 184 day Frequency interval be exceeded.	LCO 3.4.7 Table 4.4-4 Item 3	SR 3.4.17.3
3.5.1 M.1	Specifies a time requirement for verification of boron concentration whenever a SIT is drained.	SR 4.5.1.b	SR 3.5.1.4
3.5.2 M.1	Requires that the plant be taken to MODE 5 (where LCO is no longer applicable) if the SIT can not be returned to OPERABLE status.	LCO 3.5.1 Action a	LCO 3.5.2 Action C.1
		LCO 3.5.1 Action b	LCO 3.5.2 Action C.1
3.5.3 None	None	None	None
3.5.4 None	None	None	None
3.5.5 None	N/A	N/A	N/A
3.5.6 None	N/A	N/A	N/A
3.6.1 None	N/A	N/A	N/A
3.6.2 M.1	Requires verification that the OPERABLE air lock door is closed within one hour.	LCO 3.6.1.3 Action a.1	LCO 3.6.2 Actions A.1, B.1, C.2
		LCO 3.6.1.3 Action b	LCO 3.6.2 Actions A.1, B.1, C.2

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.6.3 M.1	Requires that leakage rate testing be performed at least once every 92 days for purge valves with resilient seals which are used to isolate penetrations with one or more purge valves exceeding the leakage limit.	LCO 3.6.1.7 Action c	LCO 3.6.3 Action D.3
3.6.3 M.2	Requires that penetration valves be closed to be considered in the isolation position.	LCO 3.6.3 Action 1.b	LCO 3.6.3 Action A
3.6.3 M.2	Requires that penetration valves be closed to be considered in the isolation position.	LCO 3.6.3 Action 1.b	LCO 3.6.3 Action C
3.6.3 M.3	Requires leakage rate testing of valves every 184 days and within 92 days after opening.	SR 4.6.1.7.2	SR 3.6.3.6
3.6.4 None	N/A	N/A	N/A
3.6.5 None	N/A	N/A	N/A
3.6.6 None	N/A	N/A	N/A
3.6.7 M.1	Does not allow the option of using the Hydrogen Purge System as a replacement for an inoperable hydrogen recombiner for an indefinite period of time.	LCO 3.6.4.2 Action	LCO 3.6.7 Action
		LCO 3.6.4.3 Action	LCO 3.6.7 Action
3.7.1 M.1	Places the same OPERABILITY requirements on the safety devices of both steam generators whether that specific steam generator is in operation or not.	LCO 3.7.1.1 Action b	LCO 3.7.1 Action B
3.7.1 M.2	Requires that as-left settings of each MSSV be in compliance within ± 1% of value specified in ITS Table 3.7.1-2.	SR 4.7.1.1	SR 3.7.1.1
3.7.1 M.3	Requires that if inoperable MSSV are not restored to OPERABILITY or if the RTP and VOPT are not reduced within 4 hours, the unit be placed in a MODE where the LCO does not apply.	LCO 3.7.1.1 Action a	LCO 3.7.1 Action B.2

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.7.1 M.4	Limits the amount of time that the plant can operate in MODE 3 to 12 hours before going to MODE 4 if more than four MSSVs are inoperable on one steam generator.	LCO 3.7.1.1 Action b	LCO 3.7.1 Action B.2
3.7.2 M.1	Requires that inoperable MSIVs be closed within 4 hours and verified closed once per seven days.	LCO 3.7.1.5 Action a	LCO 3.7.2 Action C
3.7.3 M.1	Requires verification of penetration isolation for inoperable MFIV once per seven days.	LCO 3.6.3 Action 1.b	LCO 3.7.3 Action A
3.7.3 M.2	Requires that with two MFIVs in the same flow path inoperable, the affected flow path be isolated within eight hours and the inoperable MFIV verified closed or isolated once per seven days.	LCO 3.6.3 Actions	LCO 3.7.3 Action B
3.7.3 M.3	Specifies limiting stroke times for MFIVs to be ≤ 9.6 seconds on an actual or simulated actuation signal while testing in accordance with the IST program.	SR 4.6.3.5	SR 3.7.3.1
3.7.4 M.1	Reduces the time allowed to restore one ADV to Operability (from 72 to 24 hours) when no ADVs are OPERABLE.	LCO 3.7.1.6	LCO 3.7.4 Action B
3.7.4 M.2	Requires that the unit be placed in MODE 4 without reliance upon steam generators for heat removal within 24 hours of entering the Condition (i.e. the following 18 hours) instead of just placing the unit in MODE 3 within 6 hours.	LCO 3.7.1.6	LCO 3.7.4 Action C
3.7.5 M.1	Requires verification (via measurement of pump d/p at a reference flow rate) that the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	SR 4.7.1.2.b.1	SR 3.7.5.2
3.7.5 M.2	Requires testing of the turbine driven AFW pump within 72 hours of reaching test conditions.	SR 4.7.1.2.b.1	SR 3.7.5.2 SR 3.7.5.3 SR 3.7.5.4

Discussion of Change	Summary of Change	CTS Section	ITS Section
		SR 4.7.1.2.e	SR 3.7.5.2 SR 3.7.5.3 SR 3.7.5.4
3.7.5 M.3	Adds an Action to require that with one steam supply to the turbine driven auxiliary feedwater pump inoperable, restoration of the steam supply system to OPERABLE status must be complete within 7 days and 10 days from discovery of the failure to meet the LCO.	LCO 3.7.1.2	LCO 3.7.5 Action A
	Require that with one AFW train inoperable for reasons other than Condition A (of ITS 3.7.5) in MODES 1, 2, or 3, restoration of the AFW train to OPERABLE status must be complete within 10 days of discovery of the failure to meet the LCO.	LCO 3.7.1.2 Action a	LCO 3.7.5 Action B
3.7.5 M.4	Removes requirement to do flow testing on a Staggered Test Basis, requiring testing prior to entering MODE 2 whenever the unit has been in MODE 5 or 6 for more than 30 days.	SR 4.7.1.2.d	SR 3.7.5.5
3.7.6 M.1	Deletes previous MODE 4 exclusion of LCO when cooldown is in progress.	LCO 3.7.1.3	LCO 3.7.6
3.7.6 M.2	Imposes an end-state requirement of "MODE 4 without reliance on steam generator for heat removal" in situations where the Condensate Storage Tank is inoperable and the backup water supply (RWMT) is also inoperable.	LCO 3.7.1.3 Action a	LCO 3.7.6 Action B
3.7.7 M.1	Extends the testing scope of automatic valves that service safety related equipment to all in the flow path.	SR 4.7.3.b	SR 3.7.7.2
3.7.8 M.1	Requires testing of the automatic actuation function of ESPS pumps	SR 4.7.4	SR 3.7.8.2
3.7.9 None	N/A	N/A	N/A
3.7.10 M.1	Not Used	N/A	N/A

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.7.10 M.2	Requires that the proper actuation of each EC System component be verified upon receipt of an actual or simulated test signal at 18 month intervals.	SR 4.7.6	SR 3.7.10.2
3.7.11 M.1	Increases scope of CREFS LCO MODE applicability to include "during movement of irradiated fuel assemblies."	LCO 3.7.7	LCO 3.7.11
3.7.11 M.2	Includes Actions for the condition "during movement of irradiated fuel assemblies."	LCO 3.7.7	LCO 3.7.11 Action D
3.7.11 M.3	Requires suspending movement of irradiated fuel assemblies.	LCO 3.7.7 Action b	LCO 3.7.11 Action E
3.7.12 M.1	Increases scope of CREATCS MODE applicability to include "during movement of irradiated fuel assemblies."	LCO 3.7.7	LCO 3.7.12
3.7.12 M.2	Requires verification at 18 month intervals that each CREATCS train has the capability to remove the assumed heat load.	SR 4.7.7.d.4	SR 3.7.12.1
3.7.12 M.3	Adds requirement to suspend movement of irradiated fuel assemblies.	LCO 3.7.7 Action b	LCO 3.7.12 Action D
		LCO 3.7.7 Action b	LCO 3.7.12 Action E
3.7.12 M.4	Includes Actions for the condition "during movement of irradiated fuel assemblies."	LCO 3.7.7	LCO 3.7.12 Action D
3.7.13 M.1	Not Used	N/A	N/A
3.7.13 M.2	Requires verification that the ESF PREACS is capable of maintaining the lower levels of the auxiliary building at a measurable negative pressure at 18 month intervals on a STAGGERED TEST BASIS.	SR 4.7.8	SR 3.7.13.4
3.7.14 None	N/A	N/A	N/A
3.7.15 None	N/A	N/A	N/A
3.7.16 M.1	Requires that Dose Equivalent I-131 be verified to be within limits at 31 day intervals.	SR 4.7.1.4 Table 4.7-1	LCO 3.7.16

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.8.3 M.1 (cont.)	Adds a Require Action for starting air receiver pressure.	~~LCO 3.8.1.3~~ Action	LCO 3.8.2 Action E
	Adds a Surveillance Requirement for lube oil inventory.	~~SR 4.8.1.3.1/2~~	SR 3.8.3.2
~~3.8.3 M.1~~	Adds a Surveillance Requirement for starting air receiver pressure.	~~SR 4.8.1.3.1/2~~	SR 3.8.3.4
3.8.3 M.2	Specifies Action Requirements based on stored diesel fuel particulates rather than on viscosity and reduces response period to 7 days.	LCO 3.8.1.3.1 Action b	LCO 3.8.3 Action C
	Specifies Action Requirements for new diesel fuel with properties outside of established limits.	LCO 3.8.1.3.1 Action c	LCO 3.8.3 Action D
	Imposes a 92 day Surveillance for diesel fuel particulate properties.	SR 4.8.1.3.1.2	SR 3.8.3.3
3.8.3 M.3	Requires that each diesel fuel oil storage tank be checked for accumulated water and removal of any found.	SR 4.8.1.3.~~1/2~~ 1 SR 4.8.1.3.1.2	SR 3.8.3.5
3.8.4 M.1	Adds an additional battery testing frequency requirement to perform a battery performance discharge test at least every 24 months when the battery has reached 85% of the expect life with capacity > 100% of manufacturer's rating.	SR 4.8.2.1.e	SR 3.8.4.8
		SR 4.8.2.1.f	SR 3.8.4.8
3.8.4 M.2	Not Used	N/A	N/A
3.8.4 M.3	Not Used	Not Used	Not Used
3.8.4 M.4	Adds restriction that the surveillance shall not be performed on MODES 1, 2, 3, and 4 on the charger credited for OPERABILITY.	SR 4.8.2.1.c.4	SR 3.8.4.6 Note
3.8.4 M.5	Adds a requirement that the battery charger be restored to OPERABLE within 24 hours when the battery has been verified to meet Category A limits.	LCO 3.8.2.1 Action b	LCO 3.8.4 Action C

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.7.17 M.1	Specifies requirements for the allowed storage locations of spent fuel assemblies within the spent fuel storage racks depending on initial enrichment and existing burnup.	Design Features 5.3 (No LCO Existed)	LCO 3.7.17
3.8.1 M.1	Requires that load sequencers be OPERABLE for Trains A and B.	LCO 3.8.1.1	LCO 3.8.1.c
	Specifies Required Actions for inoperable load sequencers.	LCO 3.8.1.1 Action	LCO 3.8.1 Action F
3.8.1 M.2	Requires a check and removal of water for diesel fuel day tanks every 92 days.	~~SR 4.8.1.1.1/2~~ LCO 3.8.1.1	SR 3.8.1.5
3.8.1 M.3	Adds requirement that voltage be ≥ 3740 V and frequency be ≥ 58.8 Hz in ≤ 10 seconds.	SR 4.8.1.1.2.d.4	SR 3.8.1.12.a/b
	Adds requirement that permanently connected loads remain energized from the offsite power system.	SR 4.8.1.1.2.d.4	SR 3.8.1.12.d
	Adds requirement that emergency loads are energized (auto-connected through the automatic load sequencer) from the offsite power system.	SR 4.8.1.1.2.d.4	SR 3.8.1.12.e
3.8.1 M.4	Removes option to only use LOP or LOCA signals as single sources to initiate the surveillance to verify that all automatic DG trips are bypassed during emergency operation.	SR 4.8.1.1.2.d.6	SR 3.8.1.13
3.8.1 M.5	Specifies a diesel start "from standby condition."	SR 4.8.1.1.2.a.2	SR 3.8.1.2
3.8.1 M.6	Not Used	N/A	N/A
3.8.2 M.1	Adds "During movement of irradiated fuel assemblies" to the LCO applicability.	LCO 3.8.1.2 Applicability	LCO 3.8.2 Applicability
3.8.2 M.2	Adds a Required Action to "immediately declare affected required feature(s) inoperable with no offsite power available."	LCO 3.8.1.2 Action	LCO 3.8.2 Action A.1
3.8.3 M.1	Adds a Require Action for lube oil inventory	LCO 3.8.1.3.1 Action	LCO 3.8.2 Action B 3

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.8.4 M.6	The ITS definition of battery degradation is changed such that degradation is indicated when the battery capacity drops by more than 10% relative to the capacity on the previous performance test rather than the average of previous tests. This is more restrictive when the results of the previous test are higher than the average of the previous performance tests.	SR 4.8.2.1.f	SR 3.8.4.8 Bases
3.8.5 M.1	Adds "During movement of irradiated fuel assemblies" to the MODE applicability.	LCO 3.8.2.2	LCO 3.8.5
3.8.5 M.2	Not Used	N/A	N/A
3.8.5 M.3	Not Used	N/A	N/A
3.8.5 M.4	Adds a requirement that the battery charger be restored to OPERABLE within 24 hours when the battery has been verified to meet Category A limits	LCO 3.8.2.2 Action b	LCO 3.8.5 Action B
3.8.6 M.1	Changes the LCO for DC Sources to that in ITS LCO 3.8.6 to be "Battery cell parameters for the Train A and Train B batteries shall be within the limits of Table 3.8.6-1." Table 3.8.6-1 lists the battery surveillance requirements. This LCO combines battery cell paramters and limits in ITS LCO 3.8.6.	LCO 3.8.2.1	LCO 3.8.6
	Changes the DC Sources Applicability to state "when associated DC electrical power subsystems are required to be Operable. This supports the distribution subsystems required by ITS LCO 3.8.10 and to include fuel handling in all MODES. Also more than one of the DC electrical power subsystems may be required in MODES 5 and 6 since the DC sources Applicability has been changed.	LCO 3.8.2.1	LCO 3.8.6 Applicability
	Adds a NOTE to the Actions stating "Separate Condition entry is allowed for each battery."	LCO 3.8.2.1	LCO 3.8.6 Actions Note

Discussion of Change	Summary of Change	CTS Section	ITS Section
	Adds verification of all connected cells instead of just pilot cells and ensures that Category C limits are met via periodic verification. Also provides for immediately declaring the battery inoperable if temperature is < 60°F. or if any Category C limits are not met.	LCO 3.8.2.1 Action b	LCO 3.8.6 Action A
	Provides for immediately declaring the battery inoperable if temperature is < 60°F. or if any Category C limits are not met.	LCO 3.8.2.1 Action b	LCO 3.8.6 Action B
3.8.6 M.2	Places additional requirements on using battery charge current of < 2 amps as a temporary substitute for specific gravity.	LCO 3.8.2.1 Table 4.8-2	LCO 3.8.6 Table 3.8.6-1
3.8.7 M.1	Only allows one inverter to be disconnected from its associated DC bus for the purpose of performing an equalizing charge on the associated battery.	LCO 3.8.3.1	LCO 3.8.7
3.8.7 M.2	Requires verification of proper inverter frequency output in addition to proper voltage.	SR 4.8.3.1	SR 3.8.7.1
3.8.8 M.1	Adds "During movement of irradiated fuel assemblies" to the MODE applicability (with a NOTE that makes Action A.2.3 not applicable in operational MODES 1, 2, 3, and 4).	LCO 3.8.3.2	LCO 3.8.8
3.8.8 M.2	Requires verification of proper inverter frequency output in addition to proper voltage.	SR 4.8.3.2	SR 3.8.8.1
3.8.9 M.1	Imposes an additional Completion Time for each Condition that requires Operability restoration within 16 hours of failure to meet the LCO.	LCO 3.8.3.1 Actions a, b, c	LCO 3.8.9 Actions A, B, C
3.8.9 M.2	Adds an additional Action E to enter LCO 3.0.3 immediately if two or more inoperable distribution subsystems result in a loss of a required safety function.	LCO 3.8.3.1 Action	LCO 3.8.9 Action E
3.8.10 M.1	Imposes an additional Required Action that states "Declare associated required shutdown cooling subsystem(s) inoperable and not in operation."	LCO 3.8.3.2 Action	LCO 3.8.10 Action A.2.5

Discussion of Change	Summary of Change	CTS Section	ITS Section
3.8.10 M.2	Adds "During movement of irradiated fuel assemblies." to the MODE Applicability.	LCO 3.8.3.2	LCO 3.8.10
3.9.1 None	N/A	N/A	N/A
3.9.2 M.1	Imposes an additional Action for both required Source Range monitors (SRM) inoperable that stipulates immediate action be taken to restore one SRM to operable status.	LCO 3.9.2 Action b	LCO 3.9.2 Action B.1
3.9.2 M.2	With both SRMs inoperable, adds the requirement to determine the boron concentration of the refueling canal within 4 hours and at least once per 12 hours thereafter.	LCO 3.9.2 Action b	LCO 3.9.2 Action B.2
3.9.2 M.3	Requires a CHANNEL CALIBRATION of the SRMs instead of just a CHANNEL FUNCTIONAL TEST.	SR 4.9.2.b	SR 3.9.2.2
		SR 4.9.2.c	SR 3.9.2.2
3.9.3 None	N/A	N/A	N/A
3.9.4 M.1	Restricts provisions of NOTE by the statement "...provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration."	LCO 3.9.8.1	LCO 3.9.4
3.9.5 M.1	Restricts provisions of NOTE by the statement "...provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration."	LCO 3.9.8.2	LCO 3.9.5
3.9.5 M.2	Requires that both SDC loops be verified OPERABLE in addition to the CTS requirement that a SDC loop be verified in operation.	SR 4.9.8.2	SR 3.9.5.1
3.9.5 M.3	Requires that power be verified to the SDC pump that is not in operation.	SR 4.9.8.2	SR 3.9.5.2
3.9.6 M.1	Expands the MODE applicability of the LCO from just when fuel is being moved within the Reactor Vessel to whenever fuel is being moved within containment.	LCO 3.9.10	LCO 3.9.6
3.9.7 None	N/A	N/A	N/A
4.0 None	N/A	N/A	N/A

Discussion of Change	Summary of Change	CTS Section	ITS Section
5.0 M.1	Requires advance authorization of deviation from the overtime guidelines.	Section 6.2.2.C	Section 5.4.1.d
5.0 M.2	Requires procedures for TS required programs.	Section 6.8.1	Section 5.4.1.e
5.0 M.3	Requires that procedures include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.	Section 6.8.1.g	Section 5.4.1.f
5.0 M.4	Adds three new programs: 5.5.13 (Diesel Fuel Testing Program), 5.5.14 (Technical Specification [TS] Bases Control Program and 5.5.15 (Safety Functions Determination Program [SFDP])	Section 6.8.1	Section 5.5.13 Section 5.5.14 Section 5.5.15
5.0 M.5	Adds the requirement to test the ESF pump room exhaust air cleanup system in accordance with ASME N510-1980.	SR 4.7.8.d	Section 5.5.11.e

Table of PVNGS Relocated Details (LA)

ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
1.1	LA.1	1.22 (Definitions)	Definition of PLANAR RADIAL PEAKING FACTOR.	Bases	2
1.1	LA.2	1.24 (Definitions)	Definition of PROCESS CONTROL PROGRAM (PCP).	ODCM	1
1.1	LA.3	Table 1.2 (Operational Modes)	K_{eff} requirement for MODE 6.	TRM	1
1.1	LA.4	Table 1.2 (Operational Modes)	Cold leg temperature requirement for MODE 6.	TRM	1
1.1	LA.5	Table 1.2 (Operational Modes)	Cold leg temperature requirement for MODEs 1 and 2.	TRM	1
2.1	LA.1	6.7.1.a	Requirement to notify the OSRC Chairman within 24 hours of Safety Limit Violation.	QA Program description	3
2.1	LA.1	6.7.1.c	Submittal of Safety Limit Violation Report to the OSRC Chairman.	QA Program description	3
2.1	LA.2	6.7.1.b	Safety Limit Violation Report reviewed by PRB.	QA Program description	3
3.0	None	N/A	N/A	N/A	N/A
3.1.1	LA.1	3.1.1.1 (Action)	Specific values for flowrate and boron concentration.	Bases	1
3.1.1	LA.2	4.1.1.1.1	SHUTDOWN MARGIN consideration factors.	Bases	2
3.1.1	LA.3	Not Used	N/A	N/A	N/A
3.1.2	LA.1	3.1.1.2 (Action a)	Specific values for flowrate and boron concentration.	Bases	1
3.1.2	LA.1	3.1.1.2 (Action b)	Specific values for flowrate and boron concentration.	Bases	1
3.1 2	LA.2	4.1.1.2.1.e	SHUTDOWN MARGIN consideration factors.	Bases	2
3.1.2	LA.2	4.1.1.2.2 (1-6)	SHUTDOWN MARGIN consideration factors.	Bases	2

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.1.2	LA.2	4.1.1.2.3 (1-6)	SHUTDOWN MARGIN consideration factors.	Bases	2
3.1.2	LA.3	Not Used	N/A	N/A	N/A
3.1.2	LA.4	4.1.1.2.1.c	Verify predicted critical CEA position is within limits.	TRM	1
3.1.2	LA.5	4.1.1.2.1.d	Prior to initial operation <5% following fuel loading consider SHUTDOWN MARGIN factors at the Transient Insertion Limits.	TRM	3
3.1.3	LA.1	4.1.1.2.4	Reference to comparison factors for overall core reactivity balance consideration.	Bases	2
3.1.4	None	N/A	N/A	N/A	N/A
3.1.5	LA.1	3.1.3.1 Action a	Discussion of cause of CEA inoperability.	Bases	2
3.1.5	LA.2	3.1.3.2	LCO Operability details of CEA indicator channels.	Bases	3
3.1.5	LA.3	3.1.3.2 Action c	Reference to method of determining when CEA is "fully out."	Bases	3
3.1.5	LA.3	3.1.3.2 Action c *	Discussion of how to determine when CEA is "fully out."	Bases	3
3.1.5	LA.4	3.1.3.4	Specification and criteria for determining CEA drop time.	TRM	1
3.1.5	LA.5	4.1.3.4.b	Requirement to check CEA drop time following potentially impacting maintenance or modification.	Bases	3
3.1.5	LA.6	3.1.3.3	LCO and ACTION for CEA Reed Switch Position Transmitter indicator channels for CEAs not fully inserted.	TRM	3
3.1.5	LA.6	3.1.3.3 (asterisk)	Qualification that LCO is applicable in specified MODES only when reactor trip breakers are in the closed position.	TRM	3
3.1.5	LA.7	3.1.3.1 Action c.2.a	Specific details for restoring CEA group alignment.	Bases	3
3.1.6	None	N/A	N/A	N/A	N/A

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.1.7	LA.1	3.1.3.6 (LCO Note ##)	Detailed description of the CEA response to a reactor power cutback.	~~UFSAR~~ COLR	2
3.1.8	None	N/A	N/A	N/A	N/A
3.1.9	LA.1	3.10.1 Action a	Specific values for flowrate and boron concentration.	Bases	1
3.1.9	LA.1	3.10.1 Action b	Specific values for flowrate and boron concentration.	Bases	1
3.1.9	LA.2	4.10.1.3	Reactor subcriticality consideration factors.	Bases	3
3.1.10	None	N/A	N/A	N/A	N/A
3.1.11	None	N/A	N/A	N/A	N/A
3.2.1	None	N/A	N/A	N/A	N/A
3.2.2	None	N/A	N/A	N/A	N/A
3.2.3	LA.1	3.2.3 Action b.1	Action if the AZIMUTHAL POWER TILT exceeds limits (due to CEA misalignment).	TRM	3
3.2.4	None	N/A	N/A	N/A	N/A
3.2.5	None	N/A	N/A	N/A	N/A
3.3.1	LA.1	4.3.1.5	Clarification as to which auto restart codes are not included in the total restart count.	Bases	2
3.3-1	LA.1	Table 3.3-1 Action 7	Clarification as to which auto restart codes are not included in the total restart count.	Bases	2
3.3.1	LA.2	Table 3.3-1	Information regarding the number of RPS channels required to trip the reactor.	Bases	2
3.3.1	LA.3	Table 3.3-1 Notes (a) & (c)	Clarifies channel trip/bypass requirements.	Bases	3
3.3.1	LA.4	Table 3.3-1 Action 2	Requirement that a bypass or trip of the process measurement circuit bypass or trip the associated multiple functional units (with details).	Bases	3
3.3.1	LA.4	Table 3.3-1 Action 3	Requirement that a bypass or trip of the process measurement circuit bypass or trip the associated multiple functional units (with details).	Bases	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.1	LA.5	Table 3.1-1 Action 3	Clarifies channel trip/bypass requirements.	Bases	3
3.3.1	LA.6	Table 4.3-1 Note (6)	Discussion as to where to inject simulated process signals while verifying channel OPERABILITY.	Bases	3
3.3.1	LA.7	Table 4.3-1 Note (7)	Description of how to determine RCS total flow rate.	Bases	3
3.3.1	LA.8	Table 4.3-1 Note (7)	Discussion of flow measurement uncertainty.	Bases	4
3.3.1	LA.9	Table 4.3-1 Note (2)	Tolerances for the difference between the calorimetric calculated power and the CPC delta T. CPC nuclear and linear power levels.	TRM	4
3.3.1	LA.9	Table 4.3-1 Note (5)	Option to use incore detectors to determine CPC shape annealing matrix elements.	Bases	3
3.3.1	LA.10	Table 3.3-1 Action 2	Review considerations for maintaining an inoperable channel in the bypass condition.	QA Program Description	3
3.3.1	LA.11	2.2 Action	Requirement that an RPS function be declared inoperable if its setpoint is less conservative than the allowable value.	Bases	3
3.3.1	LA.12	2.2.1	Requirement that trip setpoints be set consistent with the values shown in Table 2.2-1 and includes the trip setpoint related content of Table 2.2-1.	UFSAR	3
3.3.1	LA.13	Table 2.2-1 note (4)	Clarification that the steam generator level setpoints are specified in percent of the instrument range, not percent of steam generator level.	UFSAR	1
3.3.1	LA.13	Table 2.2-1 note (9)	Clarification that the steam generator level setpoints are specified in percent of the instrument range, not percent of steam generator level.	UFSAR	1

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.1	LA.14	Table 2.2-1 note (5)	Statement that the low DNBR and High LPD trip setpoints are stored in the CPC and include measurement, calculation and processor uncertainties.	Bases	4
3.3.1	LA.15	Table 2.2-1 note (6)	Defines the terms RATE, FLOOR and BAND.	UFSAR	1
3.3.1	LA.15	Table 2.2-1 note (8)	Defines the terms RATE, CEILING and BAND.	UFSAR	1
3.3.2	LA.1	Table 3.1-1	Lists information for each of the RPS functions to show the relationship between total, operable and required-for-trip.	Bases	2
3.3.2	LA.2	Table 3.3-1 Action 2	List of channel process measurement circuits that affect multiple units. The action requires bypassing or tripping the associated multiple functional units of an inoperable or in-test process measurement circuit.	Bases	1
3.3.2	LA.2	Table 3.3-1 Action 3	List of channel process measurement circuits that affect multiple units. The action requires that all functional units affected by a bypasses/tripped channel be placed in the bypasses/tripped condition also.	Bases	1
3.3.2	LA.3	Table 3.3-1 Action 3	Statement that STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.	Bases	3
3.3.2	LA.4	Table 3.3-1 Action 2	Requirement to consider the desirability of maintaining an inoperable channel in bypass in accordance with Specification 6.5.1.6.	QA Program Description	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.3	LA.1	4.3.1.4 (a. and b.)	Detailed instructions for the test requirements and the acceptance criteria for the CEA Isolation Amplifier.	Bases	1
3.3.3	LA.2	4.3.1.5	Statement that the auto restart periodic tests restart and normal system load codes shall not be included in the auto-restart total.	Bases	3
3.3.3	LA.2	Table 3.3-1 Action 7	Statement that the auto restart periodic tests restart and normal system load codes shall not be included in the auto-restart total.	Bases	3
3.3.3	LA.3	Table 4.3-1 Note (6)	Details of how to perform a CHANNEL FUNCTIONAL test and where to inject the simulated process signals.	Bases	3
3.3.4	LA.1	Table 3.3-1 Note (f)	System configuration information regarding number of channels, what each is comprised of and how they are arranged.	Bases	2
3.3.4	LA.2	Table 3.3-1	"Channels to Trip" information for each of the RPS functions is listed in the table to show the relationship between the total number of channels, minimum required and number required for a reactor trip.	Bases	2
3.3.4	LA.3	Table 4.3-1 Note (10)	Statement requiring a CHANNEL FUNCTIONAL TEST of reactor trip breakers following maintenance or adjustment	Bases	3
3.3.5	LA.1	3.3.2	Requirement that the trip setpoints must be set consistent with the values shown in Table 3.3-4.	Bases	3
3.3.5	LA.2	3.3.2 Action a	Requirement for an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4.	Bases	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.5	LA.2	3.3.2 Action b	Requirement to take action specified in Table 3.3-3 if an ESFAS instrument channel is inoperable.	Bases	3
3.3.5	LA.3	Table 3.3-3 Items I through VII	Information regarding ESFAS functions including the number of channels available and the number required for the function.	Bases	2
3.3.5	LA.4	Table 3.3-3 Action 13	Requirement to bypass or trip listed associated functional units if a channel process measurement circuit that affects multiple functional units is inoperable or in test.	Bases	3
3.3.5	LA.4	Table 3.3-3 Action 14	Requirement to place all listed functional units affected by the bypassed/tripped channel into the bypassed/tripped condition.	Bases	3
3.3.5	LA.5	Table 3.3-3 Action 14	Statement that STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent startup and/or power operation may continue if one channel is restored to OPERABLE status and the provisions of Action 13 are satisfied.	Bases	3
3.3.5	LA.6	Table 3.3-4 Items I through VII	Trip setpoint column with specific data.	UFSAR	1
3.3.5	LA.7	Table 3.3-4 Note 2	States that the percent level specified in the table for the Steam Generator Level setpoints is the percent of the distance between the Steam Generator upper and lower level narrow range instrument nozzles.	UFSAR	1

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.5	LA.7	Table 3.3-4 Note 4	States that the percent level specified in the Table for the Steam Generator Level setpoints is the percent of the distance between the Steam Generator upper and lower level wide range instrument nozzles.	UFSAR	1
3.3.5	LA.8	Table 3.3-3 Action 13	Requirement that the desirability of maintaining bypassed inoperable channels be reviewed in accordance with Specification 6.5.1.6.	QA Program Description	3
3.3.6	LA.1	Table 3.3-3	Information regarding ESFAS functions including the number of channels available and the number required for the function.	Bases	2
3.3.6	LA.2	Table 3.3-3 Note(c)	System configuration information for Initiation Logic and Manual Trip functions	Bases	2
3.3.6	LA.2	Table 3.3-3 Note (d)	Notation "The proper two out of four combination."	Bases	2
3.3.6	LA.3	Table 4.3-2 Note (3)	List of Actuation Devices that cannot be tested at power and a list of Actuation Devices that can be partially tested at power.	UFSAR	3
3.3.7	LA.1	Table 3.3-3	Information about the Loss of Power number of Channels to Trip and the Minimum Channels Operable.	Bases	2
3.3.7	LA.2	3.3.2	Requirement that the Loss of Voltage Function and the Degraded Voltage Function have their trip setpoints set consistent with the Trip Setpoint column of Table 3.3-4.	Bases	3
3.3.7	LA.2	3.3.2 Action A	Requirement that the Loss of Voltage Function and the Degraded Voltage Function have their trip setpoints set consistent with the Trip Setpoint column of Table 3.3-4.	Bases	3
3.3.7	LA.2	Table 3.3-4	Specification of Trip Value voltages for Loss of Power.	UFSAR	1

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3.3.7	LA.3	3.3.2 Action A	Requirement that an instrumentation channel be declared inoperable if its setpoint is less conservative than the value shown in the Allowable Values column of Table 3.3-4.	Bases	3
3.3.7	LA.4	Figure 3.3-1	Figure contains information relevant to OPERABILITY determination of the Loss of Voltage relays that is also included in text form in ITS 3.3.7.3.	UFSAR	1
3.3.7	LA.5	Table 3.3-3	Footnote (e) contains clarification that there are four separate loss of voltage relays and four separate degraded voltage relays per channel.	Bases	2
3.3.8	LA.1	3.3.3.1	Deleting the phrase "...with their alarm/trip setpoints within the specified limits."	Bases	3
3.3.8	LA.2	Table 3.3-6 1.D	Details of the measurement range for each radiation monitor (RU-37 and RU-38).	UFSAR	1
3.3.9	LA.1	3.3.3.1	Deleting the phrase "...with their alarm/trip setpoints within the specified limits."	Bases	3
3.3.9	LA.2	Table 3.3-6 2.B	Details of the measurement range for each radiation monitor (RU-29 & RU-30).	UFSAR	1
3.3.9	LA.3	3.3.2	Requirement that trip setpoints be set consistent with the values shown in Table 3.3-4.	Bases	3
3.3.9	LA.4	3.3.2 Action A	Actions for an ESFAS instrumentation channel trip setpoint less conservative than the value specified	Bases	3
3.3.9	LA.5	Table 3.3-3 IX	Information to show the relationship between the total number of ESFAS function channels available, minimum required OPERABLE and minimum for ESFAS actuation (CREF).	Bases	2

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.3.9	LA.6	Table 3.3-4 IX	Specification of Allowable Values for digital instrumentation that is not subject to the drift factors inherent to analog instrumentation (CREF).	Bases	1
3.3.10	LA.1	Table 3.3-10	Specification of the minimum number of channels operable for each Function	Bases	2
3.3.10	LA.2	4.6.4.1	Details for the calibration of the Containment Hydrogen monitor.	Bases	3
3.3.10	LA.3	3.3.3	Requirement that radiation monitoring instrumentation channels shown in Table 3.3-6 have alarm/trip setpoints within the specified limits.	Bases	1
3.3.10	LA.4	Table 3.3-6 1.C	Lists of the Alarm/Trip setpoints and measurement range for each Radiation Monitor (RU-148 & RU-149).	UFSAR	1
3.3.11	LA.1	Table 3.3.9.A	Readout location information for the Remote Shutdown Instrumentation.	UFSAR	2
3.3.12	LA.1	3.1.2.7 Action a.1	Information regarding the methods to be used to determine RCS boron concentration.	Bases	3
3.3.12	LA.1	3.1.2.7 Action b.1	Information regarding the methods to be used to determine RCS boron concentration.	Bases	3
3.3.12	LA.1	3.1.2.7 Note **	Information regarding where the RCS boron sample should be obtained.	Bases	3
3.3.12	LA.2	3.1.2.7 Action b.1	Information regarding the methods to be used to determine RCS boron concentration.	Bases	3
3.4.1	None	N/A	N/A	N/A	N/A
3.4.2	None	N/A	N/A	N/A	N/A
3.4.3	LA.1	4.4.8.1.2	Reactor Vessel material irradiation Surveillance Requirements	TRM	1
3.4.3	LA.1	Table 4.4-5	Reactor Vessel Material Surveillance Program - Withdrawal Schedule	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.4.4	LA.1	3.4.1.1	The phrase "...and both reactor coolant pumps in each loop..."	Bases	3
3.4.4	LA.2	4.4.1.1	The phrase "...and circulating reactor coolant.."	Bases	3
3.4.5	LA.1	3.4.1.2.a	Specifies RCP loop numbers.	Bases	2
3.4.5	LA.1	3.4.1.2.b	Specifies components making up a loop and that one RCP per loop is required.	Bases	2
3.4.5	LA.2	4.4.1.2.2	Requires that the reactor coolant loops be "circulating reactor coolant."	Bases	3
3.4.5	LA.3	4.4.1.2.3	Specifies that "indicated wide range" level is to be used for verification of SG level. The word "indicated" is removed and the remainder of the phrase moved.	Bases	3
3.4.6	LA.1	Not Used	N/A	N/A	N/A
3.4.6	LA.2	4.4.1.3.3	Shutdown cooling minimum flow rate requirements	Bases	1
3.4.6	LA.2	4.4.1.3.3	Requirement that at least one reactor coolant loop or SDC train be "circulating reactor coolant."	Bases	3
3.4.6	LA.3	3.4.1.3	Specifies reactor coolant loop and SDC train numbers	UFSAR	2
3.4.6	LA.3	3.4.1.3 (a. through d.)	Specifies components making up a reactor coolant loop	Bases	2
3.4.6	LA.4	3.4.1.3 Footnote **	Guidance in determining SG water temperature.	Bases	3
3.4.6	LA.5	4.4.1.3.2	Specifies that "indicated wide range" level is to be used for verification of SG level. The word "indicated" is deleted and the remainder of the information is relocated	Bases	3
3.4.7	LA.1	Not Used	N/A	N/A	N/A
3.4.7	LA.2	4.4.1.4.1.2	Minimum flowrate that SDC must equal or exceed.	Bases	1

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.4.7	LA.2	4.4.1.4.1.2	Statement that at least one SDC train be verified "circulating reactor coolant."	Bases	3
3.4.7	LA.3	3.4.1.4.1.b	Specifies that "indicated wide range" level is to be used for verification of SG level. The word "indicated" is deleted and the remainder of the information is relocated	Bases	3
3.4.7	LA.4	3.4.1.4.1 Footnote ##	Guidance in determining SG water temperature.	Bases	3
3.4.8	LA.1	4.4.1.4.2	Minimum flowrate that SDC must equal or exceed.	Bases	1
3.4.8	LA.1	4.4.1.4.2	Requirement that at least one SDC train be verified "circulating reactor coolant."	Bases	3
3.4.9	LA.1	4.4.3.1.3	Tests the emergency power supply for the class 1E pressurizer heaters.	TRM	2
3.4.9	LA.2	3.4.3.1	Clarification that pressurizer level for the LCO should be "steady state" and not transitory due to plant evolutions.	Bases	3
3.4.9	LA.3	3.4.3.1	Specifies "indicated level" when defining pressurizer OPERABILITY as it pertains to pressurizer level. This information, less the word "indicated" is moved to ITS Bases.	Bases	3
3.4.10	LA.1	3.4.2.2 Footnote *	Contains maintenance information concerning the approved method for setting pressurizer safety valve lift setpoints.	Bases	3
3.4.10	LA.2	Not Used	N/A	N/A	N/A
3.4.11	LA.1	3.4.2.1 Footnote *	Contains maintenance information concerning the approved method for setting pressurizer safety valve lift setpoints.	Bases	3
3.4.12	LA.1	4.4.10.a	Requirement to verify that all manual isolation valves in the pressurizer vent path are locked in the open position.	TRM	3

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3.4.12	LA.2	4.4.10.b	Details that the method for cycling each vent valve be "from the control room."	Bases	3
3.4.12	LA.3	4.4.10	Requires that vent path Surveillances be performed "when in MODES 5 or 6."	Bases	3
3.4.13	LA.1	Not Used	N/A	N/A	N/A
3.4.13	LA.2	3.4.8.3 Action b	Requirement to reduce cold leg temperature to less than 200 degrees F with one Shutdown Cooling System suction line relief valve not OPERABLE.	Bases	3
3.4.13	LA.2	3.4.8.3 Action c	Requirement to reduce cold leg temperature to less than 200 degrees F with one Shutdown Cooling System suction line relief valve not OPERABLE.	Bases	3
3.4.13	LA.3	4.4.8.3.2	Requirement that Shutdown Cooling System suction line relief valves be verified OPERABLE every 18 months	Inservice Testing Program	3
3.4.14	LA.1	4.4.5.2.1.b	Requires monitoring the containment sump inventory and discharge every 12 hours	TRM	3
3.4.14	LA.1	4.4.5.2.1.d	Requires that the reactor head flange leakoff System be monitored every 24 hours	TRM	3
3.4.15	LA.1	3.4.5.2 Footnote *(and reference)	Contains a partial list of specific PIVs excluded from requirements.	Bases	3
3.4.15	LA.1	Table 3.4-1	Reactor Coolant System Pressure Isolation Valves.	UFSAR	1
3.4.15	LA.2	4.4.5.2.2.c	Explicitly requires a demonstration of OPERABILITY following maintenance, repair or replacement work.	Bases	3
3.4.15	LA.3	4.7.11.b	Requires that performance of the Surveillance is to be accomplished during shutdown.	Bases	3
3.4.15	LA.4	4.4.5.2.2.e	Requires testing PIVs within 72 hours following a system response to an Engineered Safety Feature Actuation Signal (ESFAS).	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.4.16	LA.1	Table 3.3-6	Particulate and gaseous radioactivity monitor (RU-1) alarm setpoint.	Bases	1
3.4.16	LA.1	Table 3.3-6	Particulate and gaseous radioactivity monitor (RU-1) measurement range.	UFSAR	1
3.4.16	LA.2	Table 3.3-6 Action 27.3	Requires preparation and submittal of a special report to the commission within 30 days outlining the action taken, cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.	TRM	3
3.4.17	LA.1	Table 4.4-4 Item 4.b	Statement that one sample is sufficient if the plant has gone through a SHUTDOWN or if transient is complete in 6 hours.	Bases	3
3.5.1	LA.1	3.5.1.a	Requirement that the isolation valves be key-locked.	Bases	3
3.5.1	LA.2	3.5.1.b	Reference to borated water volume in "cubic feet."	Bases	1
3.5.1	LA.3	3.5.1.e	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.3	3.5.1.f	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.3	3.5.1 Note **	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.3	4.5.1.a.2	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.3	4.5.1.f	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.3	4.5.1.g	Specific requirements for the operation of the SIT nitrogen vent valves.	TRM	3
3.5.1	LA.4	4.5.1.e	Requires that the RCS-SIT differential pressure alarm OPERABILITY be verified at least once per 18 months.	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.5.1	LA.5	4.5.1.d	Requires verification at least once per 18 months that each SIT isolation valve opens automatically when an actual or simulated RCS pressure signal exceeds 515 psia and upon receipt of a SIAS test signal.	TRM	3
3.5.2	LA.1	3.5.1.a	Requirement that the isolation valve be key-locked.	Bases	3
3.5.2	LA.2	3.5.1 Note	Reference to borated water volume in "cubic feet."	Bases	1
3.5.2	LA.3	3.5.1.e	Requirement that the SIT nitrogen vent valves be closed with power removed.	TRM	3
3.5.2	LA.3	3.5.1.f	Requirement that the SIT nitrogen vent valves be capable of being operated upon restoration of power.	TRM	3
3.5.2	LA.3	3.5.1 Note **	Allowance to cycle the SIT nitrogen vent valves as necessary to maintain the required cover pressure.	TRM	3
3.5.2	LA.3	4.5.1.a.2	Requirement that the SIT nitrogen vent valves be closed.	TRM	3
3.5.2	LA.3	4.5.1.f	Requirement to verify that SIT nitrogen vent valves can be opened at 18 month intervals.	TRM	3
3.5.2	LA.3	4.5.1.g	Requirement to verify that power is removed from the SIT nitrogen vent valves.	TRM	3
3.5.2	LA.4	4.5.1.e	Requires that the RCS-SIT differential pressure alarm OPERABILITY be verified at least once per 18 months and specifies the method.	TRM	3
3.5.2	LA.5	4.5.1.d	Requires 18 month interval verification that the SIT isolation valve opens automatically when an actual or simulated RCS pressure signal exceeds 515 psia and upon receipt of a SIAS test signal	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.5.3	LA.1	3.5.2 (a. through c.)	List of the specific ECCS components that comprise an ECCS subsystem/train.	Bases	2
3.5.3	LA.2	3.5.2 Action b	Requires that in the event the ECCS is actuated and injects water into the RCS, a Special Report be submitted to the NRC.	TRM	4
3.5.3	LA.3	4.5.2.c	Requires that a visual inspection of the containment be made prior to establishing containment integrity and when entries are made after integrity has been established.	TRM	3
3.5.3	LA.4	4.5.2.b.2	Method of verification that the ECCS piping is full of water (by venting the accessible high points).	Bases	3
3.5.3	LA.5	4.5.2.g.1	Requires position verification of each ECCS throttle valve position stop following completion of each valve stroking operation or maintenance of the valve when the ECCS subsystems are required to be OPERABLE.	TRM	3
3.5.3	LA.6	4.5.2.h	Requires that an ECCS flow balance test be performed following completion of modifications to the ECCS subsystems that alter flow characteristics.	TRM	3
3.5.3	LA.7	4.5.2.f (1. and 2.)	Provides flowrate and developed head details of functional testing for the ECCS (HPSI and LPSI) pumps.	Inservice Testing Program	1
3.5.3	LA.8	4.5.2.e.3	Lists the specific ECCS and Containment Spray valves that are required to be actuated.	Bases	2
3.5.3	LA.9	3.7.11	Provides the LCO, Applicability and Action requirements for the Shutdown Cooling System.	TRM	3
3.5.3	LA.9	4.7.11.a	Requires that the Shutdown Cooling System be surveilled at 18 month intervals to demonstrate the specified flowpath.	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.5.4	LA.1	3.5.3 Action b	Requires that in the event the ECCS is actuated and injects water into the RCS that a Special Report be submitted to the NRC.	TRM	3
3.5.4	LA.2	3.5.3.b	Requires that ECCS subsystems contain an OPERABLE flow path with details of the path specified.	Bases	3
3.5.5	None	N/A	N/A	N/A	N/A
3.5.6	LA.1	4.5.2.d.3	Provides specific details regarding sample size, volume of borated water, etc. for verification that a sample from a TCP basket raises the pH of the mixed solution within time constraints.	Bases	1
3.6.1	LA.1	4.6.1.1.c	Requirement to test Type B penetrations.	Containment Leakage Rate Testing Program	3
3.6.2	LA.1	3.6.1.3.a	Operational requirements for the air lock doors (closure requirements).	Bases	3
3.6.3	LA.1	Not Used	N/A	N/A	N/A
3.6.3	LA.2	4.6.1.7.2	Details of purge valve leakage rate acceptance criteria.	Containment Leakage Rate Testing Program	1
3.6.3	LA.2	4.6.1.7.3	Details of purge valve leakage rate acceptance criteria.	Containment Leakage Rate Testing Program	1
3.6.3	LA.3	4.6.3.5	Statement that valves secured in their actuated position are considered operable pursuant to the specification.	Bases	3
3.6.3	LA.3	4.6.3.5 Note *** (*)	Clarification that "secured" means "Locked, sealed, or otherwise prevented from unintentional operation."	Bases	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.6.3	LA.4	4.6.3.1	Requirement that each containment isolation valve be demonstrated OPERABLE after maintenance.	Bases	3
3.6.4	None	N/A	N/A	N/A	N/A
3.6.5	LA.1	4.6.1.5	Specifies locations where containment temperatures are to be measured for use in determining the average containment air temperature.	Bases	3
3.6.6	LA.1	3.6.2.1	Functional description of the performance of an OPERABLE containment spray system.	Bases	2
3.6.6	LA.2	4.6.2.1.a	Details of ESFAS test signals and the flow paths resulting from the correct alignment of valves.	Bases	3
3.6.6	LA.2	4.6.2.1.d.1	Details of ESFAS test signals and the automatic valve actuations that result.	Bases	3
3.6.6	LA.2	4.6.2.1.d.3	Details of ESFAS test signals used in spray pump start verification.	Bases	3
3.6.6	LA.3	4.6.1.2.e	Specifies testing details of the spray nozzles for obstructions (by blowing air or smoke through them).	Bases	3
3.6.6	LA.4	4.6.2.1.b	Details of functional testing for the containment spray pumps.	Inservice Testing Program	1
3.6.7	LA.1	4.6.4.2.a.2	Details for the performance of hydrogen recombiner system functional testing.	Bases	1
3.6.7	LA.1	4.6.4.2.b	Specific reference to recombiner instrumentation (in regard to CHANNEL CALIBRATION) and functional test requirements.	Bases	3
3.6.7	LA.2	Not Used	N/A	N/A	N/A
3.7.1	LA.1	Table 3.7-1	Information on the minimum rated capacity of the MSSVs.	UFSAR	1

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.7.1	LA.1	Table 3.7-1 Note **	Clarification of MSIV capacity rating.	UFSAR	1
3.7.1	LA.2	Table 3.7-1 Note *	Details of lift set pressure testing for MSSVs	Bases	1
3.7.2	None	N/A	N/A	N/A	N/A
3.7.3	LA.1	3.6.3 Action 1.b	Details of the means of isolation.	Bases	3
3.7.3	LA.2	4.6.3.5	Statement regarding valve operability considerations.	Bases	3
3.7.3	LA.2	3.6.3 Note ***	Clarification that "secured" means "Locked, sealed, or otherwise prevented from unintentional operation."	Bases	3
3.7.4	LA.1	4.7.1.6.a	Requirement to verify the pressure in the nitrogen accumulator tank to a specific value periodically.	TRM	3
3.7.5	LA.1	3.7.1.2.a	Support requirements for steam generator auxiliary feedwater pumps (powered from separate emergency busses).	Bases	3
3.7.5	LA.1	3.7.1.2.b	Support requirements for steam generator auxiliary feedwater pump (powered from steam supply system).	Bases	3
3.7.5	LA.2	4.7.1.2.a.2	Requirement to verify the position of locked, sealed or secured manual valves in CST suction lines to AFW pumps and manual discharge valve of each pump.	TRM	3
3.7.5	LA.3	4.7.1.2.d	Details of flow path alignment and flow verification testing from CST to steam generators.	Bases	3
3.7.6	LA.1	3.7.1.3 Action b	Details of the verifications required to demonstrate OPERABILITY of the reactor makeup water tank flowpath to the essential auxiliary feedwater pumps (backup supply).	Bases	3
3.7.6	LA.1	4.7.1.3.2 (a. and b.)	Backup supply verification details.	Bases	2

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.7.7	LA.1	4.7.3.d	Requirement to periodically verify locked, sealed or secured valves to be in their correct position.	TRM	3
3.7.8	LA.1	4.7.4.2	Requirement to periodically verify locked, sealed or secured valves to be in their correct position.	TRM	3
3.7.9	None	N/A	N/A	N/A	N/A
3.7.10	LA.1	3.7.6 Action b	Completion times to determine OPERABILITY	Safety Functions Determination Program	1
3.7.10	LA.2	4.7.6.2	Requirement to periodically verify locked, sealed or secured valves to be in their correct position.	TRM	3
3.7.11	LA.1	4.7.1.1.d.2	Design details of the CREFS/SIAS actuation flowpath.	Bases	1
3.7.12	LA.1	3.7.12	Verification requirement that the control room air temperature is less than or equal to 80° F (LCO, Action and Surveillance Requirement).	TRM	3
3.7.13	LA.1	4.7.8.b	Interval and conditional requirement to demonstrate OPERABILITY of the ESF pump room air exhaust cleanup system.	Ventilation Filter Testing Program	3
3.7.14	LA.1	3.9.11 Action	Requirement to suspend "... crane operations with loads" in the fuel storage areas.	TRM	3
3.7.15	None	N/A	N/A	N/A	N/A
3.7.16	LA.1	Table 4.7-1	Requirement to perform a gross activity determination periodically.	TRM	3
3.7.17	None	N/A	N/A	N/A	N/A
3.8.1	LA.1	Not Used	N/A	N/A	N/A
3.8.1	LA.2	Not Used	N/A	N/A	N/A
3.8.1	LA.3	Not Used	N/A	N/A	N/A
3.8.1	LA.4	Not Used	N/A	N/A	N/A

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3.8.1	LA.5	Not Used	N/A	N/A	N/A
3.8.1	LA.6	3.8.1.1 Action f	Detailed information about the minimum voltages required for each unit and the number of startup transformers in service for the electrical distribution system input voltages to be within their limits.	Bases	1
3.8.1	LA.6	3.8.1.1 Action g	Detailed information about the minimum voltages required for each unit and the number of startup transformers in service for the electrical distribution system input voltages to be within their limits.	Bases	1
3.8.2	LA.1	3.8.1.2 Action 4	Requirement to suspend crane operation with loads over the fuel storage pool with less than the minimum required AC electrical power sources OPERABLE.	TRM	3
3.8.3	LA.1	Not Used	Not Used	Not Used	N/A
3.8.4	LA.1	4.8.2.1.f	Detailed information describing what constitutes a degraded battery.	Bases	3
3.8.4	LA.2	Table 3.8-1	Table names the components that comprise a DC train.	Bases	2
3.8.4	LA.3	Not Used	N/A	N/A	N/A
3.8.5	LA.1	Not Used	N/A	N/A	N/A
3.8.6	LA.1	Not Used	N/A	N/A	N/A
3.8.7	LA.1	3.8.3.1	Detailed information that describes what buses and components constitute an OPERABLE Electrical Distribution System.	Bases	2
3.8.7	LA.1	4.8.3.1	Reference to the "required manner" for energizing specified busses.	Bases	3
3.8.8	LA.1	Not Used	N/A	N/A	N/A

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
3.8.9	LA.1	3.8.3.1	Detailed information that describes what buses and combination of buses and components constitute an OPERABLE electrical distribution subsystem.	Bases	2
3.8.9	LA.1	4.8.3.1	Reference to the "required manner" for energizing specified busses.	Bases	3
3.8.9	LA.2	Not Used	N/A	N/A	N/A
3.8.10	LA.1	3.8.3.2	Detailed information that describes what buses and combination of buses and components constitute an OPERABLE electrical distribution subsystem.	Bases	2
3.8.10	LA.1	4.8.3.2	Reference to the "required manner" for energizing specified busses.	Bases	3
3.9.1	LA.1	3.9.1 Action	Details regarding boration flowrate and solution concentration.	Bases	1
3.9.1	LA.2	4.9.1.1	Requirement that the boron concentration be determined to be within the limits specified in the COLR prior to unbolting or removing the reactor pressure vessel head, and prior to withdrawal of any full length CEA.	TRM	3
3.9.2	LA.1	3.9.2	Details of startup channel neutron flux monitoring systems.	Bases	2
3.9.3	LA.1	4.9.4	Requirements for containment building penetrations to be verified in the required status prior to entering the Applicability.	TRM	3
3.9.4	LA.1	3.9.8.1 Note *	Limitation on the reasons why shutdown cooling may be removed from operation for up to one hour per eight hour period.	Bases	3
3.9.4	LA.2	4.9.8.1	Specific value for shutdown cooling loop flowrate.	Bases	1
3.9.5	LA.1	4.9.8.2	Specific value for shutdown cooling loop flowrate.	Bases	1

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3.9.5	LA.2	3.9.8.2 Note *	Limitation on the reasons why shutdown cooling may be removed from operation for up to one hour per eight hour period.	Bases	3
3.9.6	None	N/A	N/A	N/A	N/A
3.9.7	None	N/A	N/A	N/A	N/A
4.0	None	N/A	N/A	N/A	N/A
5.0	LA.1	6.1.2	Reference to Table 6.2-1	UFSAR	2
5.0	LA.1	6.2.2.a	Reference to Table 6.2-1.	UFSAR	2
5.0	LA.1	Table 6.2-1	Minimum shift crew requirements and limitations on unmanned crew positions.	UFSAR	2
5.0	LA.2	6.2.2.d	Requirement that all Core alterations be supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling.	UFSAR	3
5.0	LA.3	6.2.2.e	Site Fire Team Requirements.	UFSAR	2
5.0	LA.3	6.2.2.e Note *	Reference to Site Fire Team composition.	UFSAR	2
5.0	LA.4	6.2.3 (and Note *)	Requirements for Independent Safety Engineering Department (ISE).	QA Program Description	2
5.0	LA.5	6.4.1	Training Requirements.	UFSAR	2
5.0	LA.6	6.5.1	Requirements for the Plant Review Board (PRB).	QA Program Description	2
5.0	LA.6	6.5.2 (6.5.2.1 through 6.5.2.4 and 6.5.2.8)	Requirements for Technical Review and Control (except for modification approval requirements by Department Leader, Operations).	QA Program Description	2
5.0	LA.6	6.5.3	Requirements for the Offsite Safety Review Committee (OSRC)	QA Program Description	2
5.0	LA.7	6.6	Specific actions for reportable events.	UFSAR	3
5.0	LA.8	6.8.1.g Note (1)	Requirement for PRB approval of modifications to CPC addressable constants.	QA Program Description	2

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5.0	LA.9	6.8.1.j	Requirement to use the guidance of Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.	QA Program Description	2
5.0	LA.10	6.8.1.n (and NOTE)	Settlement Monitoring Program Implementation.	UFSAR	2
5.0	LA.10	6.8.1.o	CEA Reactivity Integrity Program Implementation.	UFSAR	2
5.0	LA.10	6.8.1.p	Fuel Assembly Surveillance Program Implementation.	UFSAR	2
5.0	LA.11	6.8.2	Procedure review and approval requirements.	QA Program Description	2
5.0	LA.11	6.8.3	Requirements for temporary changes to procedures.	QA Program Description	2
5.0	LA.12	6.8.4	Requires programs "be audited under the cognizance of the OSRC at least once per 24 months."	QA Program Description	2
5.0	LA.13	6.8.4.b	Requires that a program be established for In-Plant Radiation Monitoring.	UFSAR	2
5.0	LA.13	6.8.4.d	Requires that a program be established for Backup Method for Determining Subcooling Margin.	UFSAR	2
5.0	LA.13	6.8.4.f	Requires that a program be established for Spray Pond Monitoring.	UFSAR	2
5.0	LA.14	6.8.4.h	Provides requirements for the Radiological Environmental Monitoring Program.	TRM	2
5.0	LA.14	6.13	Provides requirements for the Process Control Program.	QA Program Description	3
5.0	LA.14	6.15	Provides requirements for Major Changes to Radioactive Liquid, Gaseous and Solid Waste Treatment Systems.	TRM	2

Type of changes

Type 1 Details of System Design and Description Including Design Limits
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Type 4 Performance Requirements for Indication-only Instrumentation and Alarms

ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
5.0	LA.15	6.9.1.1	Details associated with "Startup Report" (initiating events).	UFSAR	3
5.0	LA.15	6.9.1.2	Details associated with "Startup Report" (address).	UFSAR	3
5.0	LA.15	6.9.1.3	Details associated with "Startup Report" (submittal).	UFSAR	3
5.0	LA.16	6.9.1.5	Requirements for the information included in the Annual Report.	TRM	2
5.0	LA.17	6.9.3	Requirements for reporting of Fire Protection Program violations.	UFSAR	2
5.0	LA.18	6.10	Requirements for record retention.	QA Program Description	2
5.0	LA.18	6.14.a	References retention requirements of 6.10.2.q.	QA Program Description	2
5.0	LA.19	6.11	Details for the Radiation Protection Program.	UFSAR	2
5.0	LA.20	6.14.b	Requires PRB review and acceptance of changes to the ODCM prior to the changes becoming effective.	QA Program Description	2
5.0	LA.21	4.0.5.a	References specific 10CFR50 and ASME Code requirements governing performance of the Inservice inspection and testing.	In Service Inspection/ In Service Testing Program	2
5.0	LA.21	4.0.5.b	References Inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable addenda.	In Service Inspection/ In Service Testing Program	2
5.0	LA.21	4.0.5.c	Reference to inspection.	In Service Inspection/ In Service Testing Program	2

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
5.0	LA.22	4.5.2.e.4	Requires verification that the total measured leakage from ECCS piping and components is less than 1 g.p.m. when pressurized to at least 40 psig.	TRM	3
5.0	LA.23	3.6.1.6	Requires that the structural integrity of the containment vessel be maintained in MODES 1 through 4 and provides specific actions if it is below the acceptance criteria.	TRM	3
5.0	LA.24	4.6.1.6.1	Provides detailed surveillance and reporting requirements for the structural integrity of the containment vessel.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.24	4.6.1.6.2	Addresses how the structural integrity of the containment vessel shall be demonstrated.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.24	4.6.1.6.3	Addresses visual inspection of structural integrity components of the containment vessel.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.24	4.6.1.6.4	Addresses exterior surface inspection of the containment vessel.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
5.0	LA.24	4.6.1.6.5	Addresses reports of any abnormal degradation of structural integrity of the containment vessel.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.24	Table 4.6-1	Table of first year tendon surveillances.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.24	Table 4.6-2	Table of lift-off force for first year U-tendons.	Pre-Stressed Concrete Containment Tendon Surveillance Program	3
5.0	LA.25	4.7.7	Details of implementing the Ventilation Filter Testing Program for the ESF Pump Room Air Exhaust Cleanup System (Broad Reference).	Ventilation Filter Testing Program	3
5.0	LA.25	4.7.8	Details of implementing the Ventilation Filter Testing Program for the Fuel Building Essential Ventilation System (Broad Reference).	Ventilation Filter Testing Program	3
5.0	LA.26	3.11.1	Details of the method for implementing the Liquid Holdup Tanks including LCO, Applicability and "Action a."	TRM	3
5.0	LA.26	4.11.1	Details of the method for implementing the Liquid Holdup Tanks (sample and analysis frequency requirements).	TRM	3

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ITS NUMBER	LA Number	CTS Number	Description	Destination Document	Type of Change
5.0	LA.26	3.11.3	Details of the method for implementing the Gas Storage Tanks including LCO limit, Applicability and "Action a."	TRM	3
5.0	LA.26	4.11.3	Surveillance interval and activity limits for the Gas Storage Tanks.	TRM	3
5.0	LA.27	3.11.2	Maximum limit for the oxygen concentration in the waste gas holdup system, including Applicability and Action requirements.	TRM	1
5.0	LA.27	4.11.2	Surveillance requirements for oxygen in the waste gas holdup system.	TRM	3
5.0	LA.28	4.8.1.3.1.2	Requires that the diesel fuel storage tanks be sampled in accordance with ASTN-D4176-82.	UFSAR	3
5.0	LA.29	3.7.7	Reference to ANSI N509-1980 in the Surveillance Requirements.	UFSAR	2
5.0	LA.29	3.7.8	Reference to ANSI N509-1980 in the Surveillance Requirements.	UFSAR	2
5.0	LA.30	6.2.2.b	Requirement for number of licensed reactor operators in the control room.	UFSAR	2
5.0	LA.31	6.2.2.1.b	Specific working hours for plant staff.	UFSAR	2

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Table of PVNGS Relocated Details (LA)

The following summarizes the methods of control for various documents that will receive details that are relocated from the current technical specifications to the improved technical specifications.

Document	Method for Control
Containment Leakage Rate Testing Program	ITS 5.5 10CFR50 Appendix J. 10CFR50.59
Inservice Testing Program	ITS 5.5 10CFR50.55a 10CFR50.59
Offsite Dose Control Manual (ODCM)	ITS 5.5 10CFR50.59
Pre-Stressed Concrete Containment Tendon Surveillance Program	ITS 5.5 10CFR50.59
QA Program Description	10CFR50.54
Safety Function Determination Program	ITS 5.5 10CFR50.59
Technical Requirements Manual (TRM)	10CFR50.59
Technical Specifications Bases (Bases)	ITS 5.5 10CFR50.59
Updated Final Safety Analysis Report (UFSAR)	10CFR50.59
Ventilation Filter Testing Program	ITS 5.5 10CFR50.59
Core Operating Limits Report (COLR)	ITS 5.6 10 CFR 50.59

Insert

Table of PVNGS Less Restrictive Changes (L)

Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
1.0 L.1	Combines analog and bistable channel requirements instead of maintaining them as separate definitions which allows the bistable channel test signal to be injected "as close to the sensor as practicable.	Definitions 1.6	Definition: CHANNEL FUNCTIONAL TEST	None	unique
1.0 L.2	Relaxes the definition of CORE ALTERATION to specify the movement or manipulation of fuel and not "any component within the reactor pressure vessel."	Definitions 1.9	Definition: CORE ALTERATIONS	None	unique
1.0 L.3	Not Used	N/A	N/A	N/A	
1.0 L.4	Deletes the statement in CTS about CHANNEL FUNCTIONAL TEST that states "The CHANNEL FUNCTIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy."	Definitions 1.6	Definition: CHANNEL FUNCTIONAL TEST	None	unique
1.0 L.5	Allows for an "actual" or "simulated" signal to be used during CHANNEL FUNCTIONAL TESTING.	Definitions 1.6	Definition: CHANNEL FUNCTIONAL TEST	None	unique
1.0 L.6	Allows CHANNEL FUNCTIONAL TEST to be performed by any series of sequential, overlapping or total steps to all equipment covered by this definition.	Definitions 1.6	Definition: CHANNEL FUNCTIONAL TEST	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
1.0 L.6	Allows ENGINEERED SAFETY FEATURES RESPONSE TIME to be performed by any series of sequential, overlapping or total steps to all equipment covered by this definition.	Definitions 1.12	Definition: ENGINEERED SAFETY FEATURES RESPONSE TIME	None	unique
	Allows REACTOR PROTECTION SYSTEM RESPONSE TIME to be performed by any series of sequential, overlapping or total steps to all equipment covered by this definitions.	Definitions 1.27	Definition: REACTOR PROTECTION SYSTEM RESPONSE TIME	None	unique
1.0 L.7	Deletes the requirement for components tested under STAGGERED TEST BASIS to be performed in equal subintervals.	Definitions 1.33	Definition: STAGGERED TEST BASIS	None	unique
1.0 L.8	Relaxes the electrical power requirement under the definition of OPERABLE to permit the source to be either "normal" or "emergency" power available.	Definitions 1.19	Definition: OPERABLE	None	unique
2.0 None	None	N/A	N/A	N/A	
3.0 L.1	Adds LCO 3.0.6 to clarify the application of Required Actions for supported systems which are inoperable due to an inoperable support system. Requires only the support system's Actions to be entered if the support system's LCOs are not met.	LCO Applicability 3.0.1	LCO Applicability 3.0.2	None	unique
3.0 L.2	Permits MODE changes when LCO 3.0.4 and SR 4.0.4 are not met while in MODES and conditions other than MODE 1, 2, 3 and 4.	LCO Applicability 3.0.4	LCO Applicability 3.0.4	I	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.0 L.2	Permits MODE changes when LCO 3.0.4 and SR 4.0.4 are not met while in MODES and conditions other than MODE 1, 2, 3 and 4.	SR Applicability 4.0.4	SR Applicability 3.0.4	I	
3.0 L.3	Provides clarification that for Required Action Completion Times which require periodic performance on a "once per ..." basis, the frequency extension of 1.25 times the specified interval applies to each performance after the initial performance.	SR Applicability 4.0.2	SR Applicability 3.0.2	II	
3.0 L.4	Allowance to delay entering the Required Action up to 24 hours upon discovery of a missed Surveillance in cases where the Action Completion Time is > 24 hours.	SR Applicability 4.0.3	SR Applicability 3.0.3	None	unique
3.0 L.5	Adds a new LCO Applicability requirement that allows inoperable equipment to be returned to service under administrative controls to perform testing required to demonstrate ITS OPERABILITY or the OPERABILITY of other equipment to be within limits.	LCO Applicability Section 3.0	LCO Applicability 3.0.4	None	unique
3.1.1 L.1	Requires that the reactivity worth of any CEAs not capable of being inserted be accounted for in the determination of SDM and therefore doesn't require a separate conditional Surveillance.	SR 4.1.1.1.3	LCO 3.1.1	None	unique
3.1.1 L.2	Eliminates the requirement to perform a core reactivity balance in MODES 3, 4, and 5, since the ITS does require that SDM be determined.	SR 4.1.1.1.2	LCO 3.1.1	I	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.1 2 L.1	Reduces the MODE applicability of "Shutdown Margin - Reactor Trip Breakers Closed." by eliminating MODES 1 and 2. The SDM requirements are met through CEA alignment and insertion limits.	LCO 3.1.1.2	LCO 3.1.2	I	
3.1.2 L.2	Requires that the reactivity worth of any CEAs not capable of being inserted be accounted for in the determination of SDM and therefore doesn't require a separate conditional Surveillance.	SR 4.1.1.2.1a	LCO 3.1.2	None	unique
3.1.3 L.1	Allows 7 days to restore the overall core reactivity balance before requiring action to shutdown.	SR 4.1.1.2.4	LCO 3.1.2 Action A	VII	
3.1.3 L.2	Not Used	N/A	N/A	N/A	
3.1.3 L.3	Allows a 60 EFPD delay in performing the second overall core reactivity balance Surveillance following a refueling.	SR 4.1.1.2.4	SR 3.1.3.1	II	
3.1.4 L.1	Extends the period of time that the MTC must be determined to include a window of time within 7 EFPD prior to reaching the designated time.	SR 4.1.1.3.2.b	SR 3.1.4.2	VII	
3.1.5 L.1	Deletes the Action requirements to determine Shutdown Margin (SDM) when a CEA is inoperable.	LCO 3.1.3.1 Action a	LCO 3.1.5 Actions	None	unique
3.1.5 L.1	Deletes the Action requirements to determine Shutdown Margin (SDM) when a CEA is inoperable.	LCO 3.1.3.1 Action c	LCO 3.1.5 Actions	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.1.5 L.2	Extends the time allowed to restore misaligned CEAs to within their specified alignment from one hour to two hours.	LCO 3.1.3.1 Action c.1	LCO 3.1.5 Actions A, B and C	VII	
3.1.5 L.3	Deletes the requirement to exercise part length CEAs for OPERABILITY determination.	SR 4.1.3.1.2	LCO 3.1.5	None	unique
3.1.5 L.4	Adds an option to allow continued operation with an inoperable position indicator as long as the insertion limits are met and the CEA group is verified to be fully inserted.	3.1.3.2	3.1.5	V	
3.1.6 L.1	Extends the Completion Time for the Required Action to restore shutdown CEAs to within limits from one hour to two hours.	LCO 3.1.3.5 Action a	LCO 3.1.6 Action B.1	VII	
3.1.6 L.2	Changes Applicability for LCO from MODE 1 and 2 to MODE 1 and 2 with any regulating CEA not fully inserted.	LCO 3.1.3.5	LCO 3.1.6	I	
3.1.7 L.1	Reduces the restriction on operation between the long term steady state insertion limits and the transient insertion limits to 14 EFPD per 365 EFPD.	LCO 3.1.3.6 Action a.1.b	LCO 3.1.7 Action C	None	unique
3.1.8 None	None	N/A	N/A	N/A	
3.1.9 None	None	N/A	N/A	N/A	
3.1.10L.1	Eliminates the requirement to comply with the Linear Heat Rate (LHR) Specification during performance of PHYSICS TESTS	LCO 3.10.2	LCO 3.1.10	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
	Eliminates the Linear Heat Rate (LHR) Required Action during performance of PHYSICS TESTS	LCO 3.10.2 Action a	LCO 3.1.10	None	unique
	Deletes the Surveillance Requirements for LHR.	SR 4.10.2.2	LCO 3.1.10	None	unique
3.1.10L.2	Deletion of the requirement to be in MODE 3 within 6 hours if physics testing is suspended.	3.10.2	3.1.10	V	
3.1.11L.1	Eliminates the requirement to determine and verify that THERMAL POWER is maintained within the test power plateau.	SR 4.10.4.1	LCO 3.1.11	IX	
3.2.1 L.1	Eliminates the requirement to initiate corrective action within 15 minutes to restore the linear heat rate to within the LCO.	LCO 3.2.1 Action a.1	LCO 3.2.1 Action A.1	None	unique
		LCO 3.2.1 Action a.2	LCO 3.2.1 Action B.2.1	None	unique
3.2.2 None	None	N/A	N/A	N/A	
3.2.3 L.1	Extends the time available to reduce the Variable Overpower Trip Setpoint to 16 hours.	LCO 3.2.3 Action b.2	LCO 3.2.3 Action B.2	VII	
3.2.3 L.2	Requires placing the plant in a condition where the LCO does not apply if the Azimuthal Power Tilt is not within limits. CTS did not and therefore entry into 3.0.3 may have been required.	LCO 3.2.3 Actions	LCO 3.2.3 Action C	V	
3.2.4 L.1	Eliminates the requirement to initiate corrective action within 15 minutes to restore DNBR within the LCO limit.	LCO 3.2.4 Action a.1	LCO 3.2.4 Action A.1 Action B.1.2	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.2.5 L.1	Adds a note that allows the SR to be performed up to 2 hours after MODE 1 is greater than 20% RTP.	SR 4.2.7	SR 3.2.5.1	II	
3.3.1 L.1	Eliminates the requirement for the Logarithmic Power Level - High RPS trip function to be OPERABLE in MODE 1.	LCO 3.3.1 Table 3.3-1 Item B.2.a	LCO 3.3.1 Table 3.3.3-1 Item 2	I	
3.3.1 L.2	Eliminates the requirement to perform the Logarithmic Power Level - High functional test within 7 days of startup or prior to closing the RTCBs with the CEA drive system capable of rod withdrawal.	LCO 3.3.1 Table 4.3-1 Item I.B.2	LCO 3.3.1 Table 3.3.3-1 Item 2	II	
3.3.1 L.3	Adds a NOTE for CHANNEL FUNCTIONAL TEST of the Logarithmic Power Level Channels to state that it doesn't require testing in MODE 1 and allows a two hour time limit to perform the required testing after reducing thermal power and making MODE 2 entry.	LCO 3.3.1 Table 4.3-1 Item I.B.2	SR 3.3.1.7	II	
3.3.1 L.4	Raises to 20% the RTP at which the linear power levels, CPC delta T power and CPC nuclear power signals are adjusted to agree with the calorimetric calculation.	LCO 3.3.1 Table 4.3-1 Note (2)	SR 3.3.1.4	II	
3.3.1 L.5	Not Used	N/A	N/A	N/A	
3.3.1 L.6	Extends the time allowed to repair the INOPERABLE channel until prior to the next MODE 2 entry following the next MODE 5 entry.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.1 Action A.2 Action C.2.2	VII	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.1 L.7	Wording changed regarding instrument and bypass removal channels to clarify that only the automatic bypass removal function affects the OPERABILITY of the channel.	LCO 3.3.1	LCO 3.3.1	IX	
3.3.1 L.8	Provides a new Action for inoperable channel operational bypass removal functions for channels that are equipped with operational bypasses.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.1 Action C	IX	
3.3.1 L.9	Reduces the FUNCTIONAL TEST requirements following a CPC cabinet high temperature alarm to only the channel that alarmed.	SR 4.3.1.6	LCO 3.3.1 Action E	V	
3.3.2 L.1	Allows operation with one Logarithmic Power Level - High channel tripped and one channel bypassed longer than 48 hours and permits the plant to change MODES.	LCO 3.3.2 Table 3.3-1 Item B.2.a (Action 2)	LCO 3.3.2 Action B	IX	
3.3.2 L.2	Extends the time allowed to repair the INOPERABLE channel until prior to the next MODE 2 entry following the next MODE 5 entry.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.2 Action A.1 Action C	VII	
3.3.2 L.3	Eliminates the requirement to perform the Logarithmic Power Level - High functional test within 7 days of startup or prior to closing the RTCBs with the CEA drive system capable of rod withdrawal.	LCO 3.3.1 Table 4.3-1 Item I.B.2	LCO 3.3.2 Table 3.3.2-1 Item 1 (SR 3.3.2.2)	II	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.2 L.4	Wording changed regarding instrument and bypass removal channels to clarify that only the automatic bypass removal function affects the OPERABILITY of the channel.	LCO 3.3.1	LCO 3.3.2	IX	
3.3.2 L.5	Provides a new Action for inoperable channel operational bypass removal functions for channels that are equipped with operational bypasses.	LCO 3.3.1 Table 3.3-1 Action 2	LCO 3.3.1 Action C	IX	
3.3.2 L.6	Clarifies the intent of the requirement to lower the setpoint for the Logarithmic Power Level - High RPS trip in MODES 3, 4 and 5 with the RTCBs closed and the CEDMCS capable of CEA withdrawal. Removes the option of using the CPCs for protection in these MODES.	LCO 3.3.1 Table 3.3-1 Action 10	LCO 3.3.2 Table 3.3.2-1 Item 1	None	unique
3.3.2 L.7	Eliminates the requirement for OPERABILITY of the Low Steam Generator Pressure reactor trip function in MODE 4	LCO 3.3.1 Table 3.3-1 Item I.A.5	LCO 3.3.2 Table 3.3.2-1 Item 2 and 3	I	
3.3.2 L.7	Eliminates the requirement for OPERABILITY of the Low Steam Generator Pressure reactor trip function in MODE 4	LCO 3.3.1 Table 4.3-1 Item I.A.5	LCO 3.3.2 Table 3.3.2-1 Item 2 and 3	I	
3.3.3 L.1	Reduces the FUNCTIONAL TEST REQUIREMENT for CPCs following a cabinet high temperature alarm to only the affected channel (B or C).	SR 4.3.1.6	LCO 3.3.3 Action C	VI	
3.3.3 L.2	Extends the (INOPERABLE CEACs) Completion Time allowed to increase the DNBR margin and the Reactor Power Cutback (RPC) System to be disabled from one hour to four hours.	LCO 3.3.1 Table 3.3-1 Action 6.b.1	LCO 3.3.3 Action B	VII	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.3 L.3	Eliminates the Required Action to reduce the Surveillance Frequency from twelve hours to four hours for CEA verification Surveillances (CTS 4.1.3.5, 4.1.3.6 and 4.1.3.7).	LCO 3.3.1 Table 3.3-1 Action 6.b.3	LCO 3.3.3 Action B.2	None	unique
3.3.3 L.4	Eliminates specification of the allowed modes of movement used for CEA motion permitted by the Action Statement.	LCO 3.3.1 Table 3.3-1 Action 6.b.2.c	LCO 3.3.3 Action B.2	None	unique
3.3.4 L.1	Extends the Required Action Completion time for opening all RTCBs from one hour to six hours under the condition of one Matrix Channel inoperable in MODES 3, 4 and 5 with the RTCBs closed and the CEA system capable of CEA withdrawal (following a failure to restore the inoperable channel within 48 hours).	LCO 3.3.1 Table 3.3-1 Action 9	LCO 3.3.4 Action E	VII	
3.3.4 L.2	Extends the range of MODES in which the RTCBs may be closed for Surveillance Testing to all MODES instead of just MODES 1 and 2.	LCO 3.3.1 Table 3.3-1 Action 5	LCO 3.3.4 Action C	IX	
3.3.4 L.3	Adds a new Action for two channels of RTCBs, Manual Trip or Initiation Logic, affecting the same trip leg inoperable, that requires the affected RTCBs to be opened immediately. Because of the new Action, an entry into LCO 3.0.3 is not required for this condition.	LCO 3.3.1 Table 3.3-1	LCO 3.3.4 Action D	V VIII	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.4 L.4	Allows opening the redundant RTCB in the affected trip path with an inoperable Initiation Logic channel, Manual Trip channel or RTCB, thereby insuring trip availability while preserving failure evidence.	LCO 3.3.1 Table 3.3-1 Action 5	LCO 3.3.4 Action B	IX	
3.3.5 L.1	Extends the time allowed to repair the INOPERABLE channel until prior to the next MODE 2 entry following the next MODE 5 entry.	LCO 3.3.1 Table 3.3-3 Action 13	LCO 3.3.5 Action A	VII	
3.3.5 L.2	Provides a new Action for inoperable channel operational bypass removal functions for channels that are equipped with operational bypasses.	LCO 3.3.2 Table 3.3-3 Action 14.b	LCO 3.3.5 Action D	~~None~~ IX	unique
3.3.5 L.3	Relaxes the requirement for the Main Steam Isolation Signal (MSIS) Function (SG Press-Low, SG Press-High and Containment Press-High Signals) to be OPERABLE when all associated valves isolated by the MSIS are closed since the function is not needed in that condition.	LCO 3.3.2 Table 3.3-3 Item IV.A	LCO 3.3.5 Table 3.3.5-1 Note (c)	IX	
3.3.5 L.4	Wording changed regarding instrument and bypass removal channels to clarify that only the automatic bypass removal function affects the OPERABILITY of the channel.	LCO 3.3.2	LCO 3.3.5	IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.5 L.5	Eliminates the requirement to demonstrate during the at-power CHANNEL FUNCTIONAL TEST that the bypass logic is OPERABLE and requires it to be done within 92 days of each reactor start-up.	SR 4.3.2.2	SR 3.3.3.5	II	
3.3.5 L.6	Eliminates the MODE 4 requirement for the SIAS and MSIS sensor/trip units to be OPERABLE.	LCO 3.3.2 Table 3.3-3 Item IV.A	LCO 3.3.5 Table 3.3.5-1 Item 4	I	
		LCO 3.3.2 Table 4.3-2 Item I.A	LCO 3.3.5 Table 3.3.5-1 Item 4	I	
3.3.6 L.1	Adds a new Action for one or more functions with two initiation logic channels or Manual Trip channels affecting the same trip leg inoperable. Because of the new Action, an entry into LCO 3.0.3 is not required for this condition.	LCO 3.3.2 Table 3.3-3 Item II.B	LCO 3.3.6 Action C	V VIII	
3.3.6 L.2	Eliminates the requirement for the CSAS, MSIS and AFAS Initiation Logic, Actuation Logic and Manual Trip to be OPERABLE in MODE 4.	LCO 3.3.1 Table 3.3-1 Items: III.B, III.C, IV.B, IV.C, VI.B, VI.C, VII.B, VII.C	LCO 3.3.6 Table 3.3.6-1	I	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.6 L.2	Eliminates the requirement for the CSAS, MSIS and AFAS Initiation Logic, Actuation Logic and Manual Trip to be OPERABLE in MODE 4.	LCO 3.3.1 Table 4.3-2 Items: III.B, III.C, IV.B, IV.C, VI.B, VI.C, VII.B, VII.C	LCO 3.3.6 Table 3.3.6-1	I	
3.3.6 L.3	Changes the Action Requirement for inoperable Initiation Logic or Manual Trip (dropping the MODE 5 transition) because the requirement for CSAS, MSIS and AFAS to be OPERABLE in MODE 4 has been eliminated.	LCO 3.3.2 Table 3.3-3 Action 12	LCO 3.3.6 Action E	V	
3.3.6 L.4	Changes the Action Requirement for inoperable Matrix Logic channel (dropping the MODE 5 transition) because the requirement for CSAS, MSIS and AFAS to be OPERABLE in MODE 4 has been eliminated.	LCO 3.3.2 Table 3.3-3 Action 17	LCO 3.3.6 Action E	V	
3.3.6 L.5	Allows an additional 16 hours for restoration of an inoperable automatic actuation logic channel	LCO 3.3.2 Table 3.3-3 Action 16	LCO 3.3.6 Action D	VII	
3.3.6 L.6	Relaxes the interval requirement for testing Actuation Logic subgroup relays to 18 month intervals while shutdown, deleting the requirement to test "...during each Cold Shutdown unless tested within the previous 62 days."	LCO 3.3.2 Table 4.3-2 Note (3)	SR 3.3.6.2	II	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.6 L.7	Adds a NOTE which eliminates the requirement for the MSIS function to be OPERABLE when all the associated valves isolated by the MSIS function are closed.	LCO 3.3.2 Table 3.3-3	LCO 3.3.6 Table 3.3.6-1 Item 5 NOTE a	IX	
3.3.6 L.8	Extends the Surveillance Interval for Actuation Logic subgroup relays from 62 days (on a STAGGERED TEST BASIS) to 9 months (on a STAGGERED TEST BASIS).	SR 4.3.2.1 Table 4.3-2 Note (1), (3)	SR 3.3.6.2	None	unique
3.3.6 L.9	Eliminates the requirement for the SIAS Matrix Logic to be OPERABLE in MODE 4.	LCO 3.3.1 Table 3.3-1 Item I.B.1	LCO 3.3.6 Table 3.3.6-1 Item 1.a	I	
		LCO 3.3.1 Table 4.3-2 Item I.B.1	LCO 3.3.6 Table 3.3.6-1 Item 1.a	I	
3.3.7 L.1	Extends the time allowed to repair the INOPERABLE channel until prior to the next MODE 2 entry following the next MODE 5 entry.	LCO 3.3.2 Table 3.3-3 Action 13	LCO 3.3. Action A	VII	
3.3.7 L.2	Adds a new Action that permits one hour to restore all but two channels of LOSS OF VOLTAGE to OPERABLE status instead of forcing LCO 3.0.3 entry.	LCO 3.3.2 Table 3.3-3 Item VIII.A	LCO 3.3.7 Action C	V VIII	
3.3.7 L.3	Extends the time permitted to transition into MODE 4 if more than two channels of Degraded Voltage relays are inoperable.	LCO 3.3.2 Table 3.3-3 Action 19	LCO 3.3.7 Action D	V	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.8 L.1	Permits suspension of CORE ALTERATIONS and suspension of the movement of irradiated fuel assemblies within containment as an alternative to closing the containment purge valves if both channels of radiation monitors are inoperable	LCO 3.3.3.1 Table 3.3-6 Action 25	LCO 3.3.8 Action C	V	
3.3.8 L.2	Extends the Surveillance Interval of the radiation monitor CHANNEL FUNCTIONAL TEST to 92 days from weekly.	SR 4.3.3.1 Table 4.3-3 Inst 1.C	SR 3.3.8.2	II	
3.3.8 L.3	Eliminates the MODE 5 and 6 specific requirements for the CPIAS radiation monitors to be OPERABLE during containment purge.	SR 4.3.3.1 Table 3.3-6 Inst 1.D	LCO 3.3.8	I	
		SR 4.3.3.1 Table 4.3-3 Inst 1.C	LCO 3.3.8	I	
3.3.8 L.4	Extends the Surveillance Interval of the CHANNEL FUNCTIONAL TEST of the radiation monitors from "prior to release" to 18 months.	SR 4.3.3.1 Table 4.3-3 Inst 1.C	SR 3.3.8.3	II	
3.3.9 None	None	N/A	N/A	N/A	
3.3.10L.1	Eliminates the 7 day shutdown requirement and extends the time available to restore a single channel of PAM instrumentation to OPERABLE status to 30 days or submittal of a report to the NRC within 14 days describing the circumstances and plan for recovery.	LCO 3.3.3.6 Table 3.3-10 Action 29	LCO 3.3.10 Action A	V	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.10L.2	Extends the time available to restore two inoperable PAM channels from 48 hours to 7 days (followed by a more precise rate of decent to MODE 4).	LCO 3.3.3.6 Table 3.3-10 Action 30	LCO 3.3.10 Action E	VII	
3.3.10L.3	Extends the time available to restore inoperable Reactor Vessel Water Level instrumentation from 7 days to 30 days for a single channel inoperable.	LCO 3.3.3.6 Table 3.3-10 Action 31	LCO 3.3.10 Action A	VII	
	Extends the time available to restore inoperable Reactor Vessel Water Level instrumentation from 48 hours to 7 days for two inoperable channels.	LCO 3.3.3.6 Table 3.3-10 Action 31	LCO 3.3.10 Action C	VII	
3.3.10L.4	Relaxes the requirement for a single inoperable Containment Hydrogen Monitor by not requiring shutdown to MODE 3 if it is not restored within 30 days.	LCO 3.6.4.1 Action a	LCO 3.3.10 Action A	V VIII	
3.3.10L.5	Extends the Surveillance Interval for the Containment Hydrogen Monitor CHANNEL CHECK from once per 12 hours to once per 31 days.	SR 4.6.4.1	SR 3.3.10.1	II	
3.3.10L.6	Extends the Surveillance Interval for the Containment Hydrogen Monitor CHANNEL CALIBRATION from 92 days on a STAGGERED TEST BASIS to 18 months.	SR 4.6.4.1	SR 3.3.10.2	II	
3.3.10L.7	Deletes the Action requirement for Radiation Monitors with trip setpoints exceeding the allowable value.	LCO 3.3.3 Action a	LCO 3.3.10	IX	
3.3.10L.8	Deletes the Surveillance Requirement for a CHANNEL FUNCTIONAL TEST on the Radiation Monitors and Containment Hydrogen Monitors.	SR 4.3.3.1	LCO 3.3.10	VI	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
		SR 4.6.4.1	LCO 3.3.10	VI	
3.3.10L.9	Deletes the requirement for the Containment Area Radiation Monitors to be OPERABLE in MODE 4.	LCO 3.3.3.1 Table 3.3-6 Item 1.C	LCO 3.3.10 Table 3.3.10-1 Item 9	I	
		LCO 3.3.3.1 Table 4.3-3 Item 1.D	LCO 3.3.10 Table 3.3.10-1 Item 9	I	
3.3.10L.10	Extends the time available to restore inoperable Containment Area Radiation Monitors from 72 hours to 30 days followed by submittal of a report to the NRC for a single channel inoperable.	LCO 3.3.3.6 Table 3.3-6 Action 27	LCO 3.3.10 Action A	VII	
	Extends the time available to restore inoperable Containment Area Radiation Monitors from 72 hours to 7 days followed by submittal of a report to the NRC for two channels inoperable.	LCO 3.3.3.6 Table 3.3-6 Action 27	LCO 3.3.10 Action C	VII	
3.3.10L.11	Extends the Surveillance Interval for the Containment Area Radiation Monitor CHANNEL CHECK from once per 12 hours to once per 31 days.	SR 4.3.3.1 Table 4.3-3 Item 1.D	SR 3.3.10.1	II	
3.3.10L.12	Eliminates the NOTE that states that the provisions of Specification 3.0.3 are not applicable. Since ITS provides direction for two Containment Area Radiation Monitors inoperable and CTS did not, this NOTE is no longer necessary.	LCO 3.3.3.1 Action c	LCO 3.3.10	V VIII	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.10L.13	Adds a NOTE to clarify that CHANNEL CALIBRATION for Neutron Detectors is not required, thus eliminating this Surveillance Requirement.	SR 4.3.3.6 Table 4.3-7 Item 16	SR 3.3.10.2	VI	
3.3.11L.1	Extends the time allowed to restore an inoperable channel from 7 days to 30 days.	LCO 3.3.3.5 Action a, b	LCO 3.3.11 Action A, B	V VIII	
3.3.11L.2	Adds the statement "that is normally energized" to identify instrumentation channels that require CHANNEL CHECKS. This statement eliminates the need to energize instrumentation (that is not normally energized at power) in order to test it and thereby reduces the scope of testing.	SR 4.3.3.5.a	SR 3.3.11.1	II	
3.3.11L.3	Surveillance wording changed to permit the use of continuity checks to verify that disconnect switches will open when required.	SR 4.3.3.5.b	SR 3.3.11.2	VI	
3.3.11L.4	Adds a NOTE to clarify that CHANNEL CALIBRATION for Neutron Detectors is not required, thus eliminating this Surveillance Requirement.	SR 4.3.3.5	SR 3.3.11.3	VI	
3.3.12L.1	Combines Required Actions for two channels of Boron Dilution Alarms inoperable and eliminates the MODE 6 related actions (MODE 6 requirements are addressed in ITS 3.9.2).	LCO 3.1.2.7 Action b.1 Action b.2	LCO 3.3.12 Action B	V	
3.3.12L.2	Not Used	N/A	N/A	N/A	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.3.12L.3	Changes the OPERABILITY requirements for MODEs 3, 4 and 5 with RTCBs open by deleting the need for two channels of Logarithmic Power Indication and adding the requirement for two channels of Boron Dilution Alarms.	LCO 3.3.1 Table 3.3-1 Item I.B.2.b	LCO 3.3.12	none	unique
		LCO 3.3.1 Table 4.3-1 Item I.B.2.b	LCO 3.3.12	none	unique
3.3.12L.4	Extends the Surveillance Interval for the BDAS CHANNEL FUNCTIONAL TEST from 31 days of cumulative operation during shutdown to 92 days and adds a NOTE stating that the SR isn't required until 72 hours after neutron flux is within the startup range.	SR 4.1.2.7.b	SR 3.3.12.2	II	
3.4.1 L.1	Introduces a NOTE which specifies two instances when the pressurizer pressure LCO doesn't apply, thus relaxing the requirements from CTS.	LCO 3.2.8	LCO 3.4.1	IX	
3.4.1 L.2	Relaxes the Action Requirement for when RCS flow rate is determined to be less than the limit by allowing two hour in which to attempt to restore flow and two additional hour in which to reach MODE 2.	LCO 3.2.5 Action	LCO 3.4.1	VII	
3.4.2 L.1	Extends the time period prior to achieving criticality in which the RCS cold leg temperature must be determined to be >545°F from 15 minutes to 30 minutes.	SR 4.1.1.4	SR 3.4.2.1	II	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.4.3 L.1	Extends the time allowed to determine RCS suitability for further operation following out of parameter conditions from 6 hours to 72 hours.	LCO 3.4.8.1 Action	LCO 3.4.3 Action A.2	VII	
3.4.4 L.1	Extends the time allowed to reach MODE 3 with less than 2 RCPs in each loop from 1 hour to 6 hours.	LCO 3.4.4.1 Action	LCO 3.4.4 Action A	VII	
3.4.5 None	None	N/A	N/A	N/A	
3.4.6 None	None	N/A	N/A	N/A	
3.4.7 L.1	Adds a NOTE that allows all SDC trains to be removed from operation during a planned heatup to MODE 4 when at least one RCS loop is in operation.	SR 4.4.1.4.1.2	LCO 3.4.7 NOTE 5	IX	
3.4.8 None	None	N/A	N/A	N/A	
3.4.9 None	None	N/A	N/A	N/A	
3.4.10L.1	Addresses the condition where two or more pressurizer safety valves are inoperable and thereby avoids a forced entry into LCO 3.0.3 as is currently required by CTS.	LCO 3.4.2.2 Actions	LCO 3.4.10 Action B	None	unique
3.4.10L.2	Permits pressurizer safety valve settings to be outside the limits of the LCO for 72 hours following entry into MODE 3 for the purpose of setting the pressurizer safety valve lift settings under ambient conditions.	LCO 3.4.2.2	LCO 3.4.10	IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.4.10L.3	Eliminates current conflict between CTS 3.4.2.2 and 3.4.2.1 regarding more than one but not all pressurizer safety valves (PSVs) OPERABLE. The requirements of CTS 3.4.2.1 are utilized to permit operation in MODE 4 with one or more PSVs OPERABLE.	LCO 3.4.2.1 Actions LCO 3.4.2.2 Actions	LCO 3.4.10 LCO 3.4.11	IX	
3.4.11L.1	Eliminates the requirement to suspend all operations involving positive reactivity changes with no pressurizer code safety valve operable.	LCO 3.4.2.1 Action a	LCO 3.4.11 Action A	V	
3.4.11L.2	Permits pressurizer safety valve settings to be outside the limits of the LCO for 72 hours following entry into MODE 3 (and in MODE 4) for the purpose of setting the pressurizer safety valve lift settings under ambient conditions.	LCO 3.4.2.1 Action b	LCO 3.4.11 Action B	IX	
3.4.11L.3	Relaxes the MODE 4 Applicability of the Pressurizer Safety Valve LCO depending on RCS cold leg temperature.	LCO 3.4.2.1	LCO 3.4.11	I	
3.4.12L.1	Relaxes the MODE 4 applicability for the Pressurizer Vents LCO to MODE 4 with pressure ≥ 385 psig.	LCO 3.4.10	LCO 3.4.11	I	
3.4.13L.1	Changes the Surveillance Frequency for the RCS vent pathway from 12 hours to 31 days since PVNGS has no valves in the RCS vent pathway.	SR 4.4.8.3.1	SR 3.4.13.1	II	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.4.13L.2	Includes the provisions for a depressurized RCS with a 16 square-inch RCS vent without requiring the unnecessary removal of the reactor vessel head. It also allows entry into a MODE of Applicability for ITS LCO 3.4.13 with one or more SCS relief valves inoperable.	LCO 3.4.8.3	LCO 3.4.13	IX	
3.4.14L.1	Eliminates the explicit requirement for early RCS water inventory balance performance.	LCO 3.4.5.2 Action d	LCO 3.4.14	V	
3.4.14L.2	Explicitly details the requirements for completing the RCS water inventory balance.	SR 4.4.5.2.1.c	SR 3.4.14.1	VI	
3.4.15L.1	Changes the limit on leakage from PIVs from 1 gpm to a rate based on the size of the valve (up to 5 gpm).	LCO 3.4.5.2	LCO 3.4.15	VI	
3.4.15L.2	Provides an exception to MODE 4 for operability of the SDC PIVs "when in, or during the transition to or from, the SDC mode of operation."	LCO 3.4.5.2	LCO 3.4.15	IX	
3.4.15L.3	Allows the use of a check valves to perform the function of isolation when RCS pressure isolation valve leakage is above the limit.	LCO 3.4.5.2 Action C	LCO 3.4.15 Action A.1	V	
3.4.15L.4	Eliminates the LCO 3.0.4 (CTS 4.0.4) exclusion footnote and modifies the SR performance requirements in its place.	SR 4.4.5.2.2	SR 3.4.15.1 NOTE	IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.4.15L.5	Adds an allowance to limit testing of PIVs actuated during performance of the Surveillance, providing that the PIV actuation is unavoidable.	SR 4.4.5.2	SR 3.4.15.1	VI	
3.4.16L.1	Adds an LCO 3.0.4 exemption to the Action Statements A and B which allow changing MODEs while RCS leakage detection instrumentation are inoperable.	LCO 3.4.5.1	LCO 3.4.16	IX	
3.4.16L.2	Extends the Completion Time for obtaining grab samples and analyzing them from 12 hours to obtain and 3 to analyze to a total of 24 hours.	LCO 3.4.5.1 Action a	LCO 3.4.16 Action B	VII	
3.4.16L.3	Eliminates the requirement to return the inoperable containment atmosphere radioactivity monitor to Operable status within 72 hours or place a moveable air monitor in-line.	LCO 3.3.3.1 Action 27	LCO 3.4.16 Action B	V	
3.4.17L.1	Eliminates the RCS Specific Activity MODE 4 and 5 Applicability and reduces the MODE 3 Applicability to $T_{cold} \geq$ 500°F.	LCO 3.4.7	LCO 3.4.17	I	
3.4.17L.2	Adds a NOTE that states that LCO 3.0.4 are not applicable, thus a MODE change is allowed with the reactor coolant specific activity Dose Equivalent I-131 is > 1.0 microcuries per gram.	LCO 3.4.7	LCO 3.4.17 Action A	IX	
3.4.17L.3	Relaxes the sample frequency from 72 hours to 7 days.	LCO 3.4.7 Table 4.4-4 Item 1	SR 3.4.17.1	II	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.4.17L.4	Reduces the Applicability of the Surveillance to verify Dose Equivalent I-131 to only require performance in MODE 1.	LCO 3.4.7 Table 4.4-4 Item 4.b	SR 3.4.17.2	I	
3.5.1 L.1	The allowed outage time for a SIT that is INOPERABLE due to boron concentration being outside of the limits has been increased from 1 hour to 72 hours.	3.5.1	3.5.1	VII	
3.5.1 L.2	The action requirements for a SIT that is INOPERABLE due to a closed isolation valve have been changed.	3.5.1	3.5.1	V	
3.5.2 L.1	Relaxes the requirement to verify that power is removed from the SIT isolation valve operator, raising it from 430 psia pressurizer pressure to 1500 psia.	SR 4.5.1.c	SR 3.4.2.5	VI	
3.5.2 L.2	The allowed outage time for a SIT that is INOPERABLE due to boron concentration being outside of the limits has been increased from 1 hour to 72 hours.	3.5.1	3.5.2	VII	
3.5.2 L.3	The action requirements for a SIT that is INOPERABLE due to a closed isolation valve have been changed.	3.5.1	3.5.2	V	
3.5.2 LB.1	Changes the minimum nitrogen cover pressure (indicated) from 254 psig to 260 psig	LCO 3.5.1 Note +	SR 3.5.2.3	None	unique
3.5.3 L.1	Relaxes the requirements for restoring ECCS trains to OPERABILITY. This permits taking credit for combining OPERABLE components from different trains to achieve the equivalent availability of 100% ECCS flow.	LCO 3.5.2 Action a	LCO 3.5.3 Action B	IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.5.3 L.2	Reduces the Actions necessary to respond to a condition where the ECCS subsystem cannot be returned to OPERABLE status. ITS requires that pressurizer pressure be reduced to less than 1837 psia and T_c be reduced to <485°F where CTS requires that the plant be placed in HOT STANDBY.	LCO 3.5.2 Action a	LCO 3.5.3 Action C	V	
3.5.3 L.3	Expands the methods allowed for initiating a signal to verify ECCS response. CTS required the use of a test signal while ITS allows using an actual or simulated signal.	SR 4.5.2.e.1 SR 4.5.2.e.2 SR 4.5.2.e.3	SR 3.5.3.4 SR 3.5.3.5 SR 3.5.3.6	VI	
3.5.3 L.4	Eliminates the Surveillance Requirement to verify that listed valves in the ECCS flow path are in their required position. The PVNGS design doesn't include any valves that meet the Bases for this Surveillance.	SR 4.5.2.a	LCO 3.5.3	VI	
3.5.4 L.1	Extends the time allowed to place the plant in COLD SHUTDOWN with no ECCS subsystem OPERABLE by four hours.	LCO 3.5.3 Action a	LCO 3.5.4 Action A	V	
3.5.5 L.1	Extends the time allowed to restore the RWT temperature to within OPERABLE status from 1 hour to 8 hours.	LCO 3.5.4 Action	LCO 3.5.5 Action A	VII	
3.5.6 L.1	Provides an 72 hour Action to restore inoperable TSP to within its limit and thus does not force an entry into Specification 3.0.3.	LCO 3.5.2	LCO 3.5.6 Action A	V	
3.6.1 None	None	N/A	N/A	N/A	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.6.2 L.1	Removes the restriction for restoring the overall airlock leakage prior to performing the next surveillance test.	LCO 3.6.1.3 Action a.1	LCO 3.6.2 Action A	V	
3.6.2 L.2	Expansion of allowance to open the OPERABLE air lock door and total time it may remain open.	LCO 3.6.1 3 Action a.1	LCO 3.6.2 Action NOTE 1 Action A NOTE 2	IX	
3.6.2 L.3	Relaxation of airlock door position verification requirement to permit it to be done by administrative means if the Operable door is in a high radiation area.	LCO 3.6.1.3 Action a.1	LCO 3.6.2 Action A.3, B.3	IX	
3.6.2 L.4	Eliminates requirement to shutdown if the airlock door is inoperable due to an inoperable interlock mechanism.	LCO 3.6.1.3 Action b	LCO 3.6.2 Action B	VII	
3.6.2 L.5	Extends the surveillance interval for airlock doors from 6 months to 24 months.	SR 4.6.1.3.b	SR 3.6.2.2	II	
3.6.3 L.1	Relaxation of position verification requirements for manual valves, blind flanges and deactivated automatic valves to permit it to be done by administrative means if the device is located in a high radiation area.	SR 4.6.1.1.a	LCO 3.6.3 Actions A.2, C.2 and E.2 SR 3.6.3.3 SR 3.6.3.4	VI IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.6.3 L.2	Extends the leak rate testing surveillance interval on 42" Containment purge valves from once per 6 months (on a STAGGERED TEST BASIS) to every 184 days and within 92 days after opening the valve.	SR 4.6.1.7.2	SR 3.6.3.6	II	
	Extends the leak rate testing surveillance interval on 8" containment purge valves from 92 days to 184 days and within 92 days after opening.	SR 4.6.1.7.3	SR 3.6.3.6	II	
3.6.3 L.3	Expands the permitted method of verification of automatic valve actuation from only using a test signal to using either a simulated or actual actuation signal.	SR 4.6.3.2	SR 3.6.3.7	VI	
3.6.3 L.4	Relaxes Surveillance Requirements for automatic CIVs to only require testing valves that are not locked, sealed or otherwise secured in position (which are administratively controlled).	SR 4.6.3.2	SR 3.6.3.7	VI	
3.6.3 L.5	Eliminates the requirement for 8" containment purge valves to be sealed closed	SR 4.6.1.7.4	SR 3.6.3.2	VI	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.6.3 L.6	Relaxes the surveillance interval for verification of penetrations not capable of being closed by OPERABLE containment isolation valves and required to be closed during accident conditions. The interval is changed from "...during each cold shutdown but not more often than once per 92 days" to "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days."	SR 4.6.1.1.a	LCO 3.6.3 Action A.2, D.2 SR 3.6.3.4	II VII	
3.6.3 L.7	Reduces the position verification requirements for containment isolation valves and blind flanges that are not capable of automatic closure and required closed during accident conditions. The scope of this verification is reduced to just those valves that are not locked, sealed or otherwise secured in the closed position.	SR 4.6.1.1.a SR 4.6.1.1.a*	SR 3.6.3.3 SR 3.6.3.4	VI	
3.6.3 L.8	Reduces the scope of stroke time testing requirements from power operated or automatic valves used in CIAS, CPIAS or CSAS to just automatic power operated containment isolation valves.	SR 4.6.3.3	SR 3.6.3.5	None	unique
3.6.3 L.9	Extends the time allowed to respond with two containment isolation valves inoperable in the same penetration, allowing an hour to isolate the affected penetration prior to initiating a shutdown.	LCO 3.6.3 Action 1.d	LCO 3.6.3 Action B	VII	
3.6.4 None	None	N/A	N/A	N/A	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.6.5 LB.1	Changes maximum containment air temperature from 120 degrees F to 117 degrees F to account for instrument uncertainty.	LCO 3.6.1.5	LCO 3.6.5	None	unique
3.6.6 L.1	Reduces the scope of valves in the flowpath required to be verified in the correct position to permit suction on the RWT on a CSAS test signal, eliminating valves that are locked, sealed or otherwise secured in position.	SR 4.6.2.1.a	SR 3.6.6.1	VI	
3.6.6 L.2	Expands the scope of initiating signals permitted to verify their automatic performance from just a test signal to either a simulated or actual initiation signal.	SR 4.6.2.1	SR 3.6.6.4 SR 3.6.6.5	VI	
3.6.6 LB.1	Changes the applicability of the containment spray specification to specify that RCS pressure must be greater than or equal to 385 psia.	LCO 3.6.2.1 Applicability	LCO 3.6.6 Applicability	None	unique
3.6.6 LB.2	Changes the containment spray fill header from 115 feet to 113 feet.	SR 4.6.2.1.c	SR 3.6.6.2	None	unique
3.6.7 L.1	Adds a condition to address the situation with two hydrogen recombiners inoperable and permits continued plant operation up to 7 days instead of forcing entry into Specification 3.0.3.	LCO 3.6.4.2	LCO 3.6.7 Condition B	V	
3.6.7 L.2	Not used	N/A	N/A	N/A	
3.6.7 L.3	Adds a NOTE which states that LCO 3.0.4 is not applicable and therefore permits MODE changes.	LCO 3.6.7	LCO 3.6.7 Action A	IX	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.1 L.1	Adds VOPT and RTP information which allows continued operation with up to 4 inoperable MSSVs per operating steam generator while still complying with the LCO.	LCO 3.7.1.1	LCO 3.7.1	IX	
3.7.1 L.2	Extends the completion time for reducing the VOPT setpoint with one or more MSSVs inoperable from 4 hours to 12 hours.	LCO 3.7.1.1 Action a	3.7.1 Action A.2	VII	
3.7.1 L.3	Deletes the requirement for MSSVs to be OPERABLE in MODE 4.	LCO 3.7.1.1	LCO 3.7.1	I	
3.7.2 L.1	Requires four MSIVs be OPERABLE and adds a NOTE that states that "Separate Condition entry is allowed for each MSIV."	LCO 3.7.1.5	LCO 3.7.2 Action C	IX	
3.7.2 L.2	Allows the flexibility of using an actual or simulated actuation signal to initiate closed stroke time testing of the MSIVs.	SR 4.7.1.5.1	SR 3.7.2.1	VI	
3.7.2 L.3	Allows entry into MODE 3 to perform Surveillance testing of the MSIVs but doesn't specify a time limit for performing the Surveillance.	SR 4.7.1.5.2	SR 3.7.2.1	II	
3.7.2 L.4	Reduces the scope of MODE applicability by eliminating the condition where MSIVs are closed while in MODE 1, 2, 3 or 4.	LCO 3.7.1.5	LCO 3.7.2	I	
3.7.3 L.1	Reduces the scope of MODE applicability by eliminating the condition where MSIVs are closed or isolated by a closed power operated valve while in MODE 1, 2, 3 or 4.	LCO 3.6.3	LCO 3.7.3	I	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.3 L.2	Extends the time available to close the inoperable MSIV or isolate the penetration from 4 hours to 72 hours.	LCO 3.6.3 Action 1.a Action 1.b	LCO 3.7.3. Action A	VII	
3.7.4 L.1	Adds a NOTE which permits changing MODEs (3.0.4 exclusion) with only one OPERABLE ADV.	LCO 3.7.1.6 Action	LCO 3.7.3 Action A	IX	
3.7.4 L.2	Extends interval for exercising ADVs from prior to startup following any cold shutdown of 30 days or longer to 18 month intervals.	SR 4.7.1.6.b	SR 3.7.4.1	II	
3.7.5 L.1	Reduces the OPERABILITY requirements for AFW trains in MODE 4 to a single Train including a motor driven pump.	LCO 3.7.1.2	LCO 3.7.5	IX	
	Changes the Applicability of the Action requirements for multiple AFW pumps inoperable to reflect MODES 1, 2 and 3.	LCO 3.7.1.2 Action b, c	LCO 3.7.5 Action C, D	IX	
	Adds a new Action and NOTE to provide direction when the required AFW pump is not OPERABLE in MODE 4.	LCO 3.7.1.2	LCO 3.7.5 Action E	IX	
3.7.5 L.2	Adds a new NOTE which states "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status" which has the effect of suspending shutdowns.	LCO 3.7.1.2 Action c	LCO 3.7.5 Action D	V	
3.7.5 L.3	Eliminates the requirement to verify the correct actuation of valves which are locked, sealed, or otherwise secured in position.	SR 4.7.1.2.c.1	SR 3.7.5.3	VI	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.5 L.4	Changes the MODE 4 Surveillance testing of AFW pump and valve automatic actuation by not requiring testing when the steam generator is required for heat removal (the pump is already running and auto-start is not required).	SR 4.7.1.2.c	SR 3.7.5.3 SR 3.7.5.4	II	
3.7.5 L.5	Changes the Surveillance to allow the use of either a simulated or actual actuation signal for Surveillance purposes, thus permitting non-Surveillance actuations to be used to fulfill Surveillance requirements.	SR 4.7.1.2.c.1 SR 4.7.1.2.c.2	SR 3.7.5.3 SR 3.7.5.4	VI	
3.7.5 L.6	Eliminates the requirement to test AFW pumps on a STAGGERED TEST BASIS.	SR 4.7.1.2.b	SR 3.7.5.2	II	
3.7.6 L.1	Extends the time available to achieve end-state conditions for Action requirements from 12 hours to 24 hours (inoperable CST with the RWMT not verified operable within 4 hours or CST level can not be restored within 7 days).	LCO 3.7.1.3 Action b	LCO 3.7.6 Action B	V	
3.7.7 L.1	Limits the requirement to verify automatic valves servicing safety related equipment actuating to their correct position to only those valves that are not locked, sealed, or otherwise secured in position.	SR 4.7.3.b	SR 3.7.7.2	VI	
3.7.8 None	None	N/A	N/A	N/A	
3.7.9 None	None	N/A	N/A	N/A	
3.7.10None	None	N/A	N/A	N/A	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.11L.1	Relaxes the MODE 5 and 6 requirement for the CREFS train placed in operation to comply with the Required Action to be capable of being powered by an OPERABLE emergency power source.	LCO 3.7.7 MODE 5 and 6 Action b	3.7.11 Action C	IX	
3.7.11L.2	Eliminates the requirement to test CREFS on a STAGGERED TEST BASIS.	SR 4.7.7.a	SR 3.7.11.1	II	
3.7.11L.3	Changes the Surveillance to allow the use of either a simulated or actual actuation signal for Surveillance purposes, thus permitting non-Surveillance actuations of CREFS to be used to fulfill Surveillance requirements.	SR 4.7.7.d:2	SR 3.7.11.3	VI	
3.7.11L.4	Eliminates the requirement to suspend positive reactivity (boron concentration) changes in MODES 5 and 6	LCO 3.7.7 MODES 5 and 6 Action b	LCO 3.7.11 Action E	IX	
3.7.11L.5	Extends the Surveillance testing interval of CREFS positive pressure testing by changing from 18 month intervals to 18 months on a STAGGERED TEST BASIS.	SR 4.7.7.d.3	SR 3.7.11.4	II	
3.7.12L.1	Relaxes the requirement for the CREATCS train placed in operation to be capable of being powered from an OPERABLE emergency power source.	LCO 3.7.7 MODES 5 and 6 Action b	LCO 3.7.12 Action C	IX	
3.7.12L.2	Eliminates the requirement to suspend positive reactivity (boron concentration) changes in MODES 5 and 6	LCO 3.7.7 MODES 5 and 6 Action b	LCO 3.7.12 Action E	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.12L.3	Extends the time available to restore the inoperable CREFS train to OPERABILITY from 7 days to 30 days.	3.7.7 Action	LCO 3.7.12 Action A	VII	
3.7.12L.4	Not Used	N/A	N/A	N/A	
3.7.13L.1	Eliminates the requirement to conduct the ESF 31 day Surveillance on a STAGGERED TEST BASIS.	SR 4.7.8.a	SR 3.7.13.1	II	
	Eliminates the requirement to initiate the ESF 31 day Surveillance from the Control Room.	SR 4.7.8.a	SR 3.7.13.1	VI	
3.7.13L.2	Changes the Surveillance to allow the use of either a simulated or actual actuation signal for Surveillance purposes, thus permitting non-Surveillance actuations of the ESF PREACS to be used to fulfill Surveillance requirements.	SR 4.7.8.d.2	SR 3.7.13.3	VI	
3.7.14L.1	Relaxes the Applicability to only during movement of irradiated fuel assemblies within the fuel storage pool. instead of the CTS "whenever irradiated fuel assemblies are in the storage pool."	LCO 3.9.11	LCO 3.7.14	I	
3.7.14L.2	Eliminates the requirement to restore water level to within its limit since suspending fuel movement precludes a fuel handing accident.	LCO 3.9.11 Actions	LCO 3.7.14	V	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.7.15L.1	Reduces the Applicability of "whenever fuel assemblies are in the spent fuel storage pool" by adding "and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool."	LCO 3.9.13	LCO 3.7.15	I	
3.7.16None	None	N/A	N/A	N/A	
3.7.17None	None	N/A	N/A	N/A	
3.8.1 L.1	Changes the Surveillances to allow the use of either a simulated or actual actuation signal for Surveillance purposes, thus permitting non-Surveillance DG starts to be used to fulfill Surveillance requirements.	SRs 4.8.1.1.2.d.3 4.8.1.1.2.d.4 4.8.1.1.2.d.5 4.8.1.1.2.d.10	SRs 3.8.1.11 3.8.1.12 3.8.1.17 3.8.1.19	VI	
3.8.1 L.2	Removes the 24 hour constraint and allows the LCO to be exited prior to completion of the common mode failure evaluation without having to run the other DG.	LCO 3.8.1.1.b Footnote 1	LCO 3.8.1 Action B.3	None	unique
3.8.1 L.3	Relaxes the constraint on performing the manual transfer of the onsite Class 1E power supply in MODES 3 and 4.	SR 4.8.1.1.1.b	SR 3.8.1.8 Note	VI	
3.8.2 None	None	N/A	N/A	N/A	
3.8.3 None	None	N/A	N/A	N/A	
3.8.4 L.1	Not Used	N/A	N/A	N/A	
3.8.4 L.2	Allows the flexibility to perform a modified performance discharge test on the battery instead of a full performance discharge.	SR 4.8.2.1.e	SR 3.8.4.8	VI	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.8.4 L.3	Deletes the requirement to verify that battery cell-to-cell and terminal connections are "tight."	SR 4.8.2.1.c.2	SR 3.8.4.4 SR 3.8.4.5	VI	
3.8.4 L.4	The ITS definition of battery degradation is changed such that degradation is indicated when the battery capacity drops by more than 10% relative to the capacity on the previous performance test rather than the average of previous tests. This is less restrictive when the results of the previous test are lower than the average of the previous performance tests.	4.8.2.1.f	SR 3.8.4.8 Bases	II	
3.8.5 L.1	Not Used	N/A	N/A	N/A	
3.8.5 L.2	Adds a NOTE which identifies SRs precluded to prevent the DC sources necessary to support the DC electrical power subsystem(s) required by LCO 3.8.10 from being discharged or otherwise rendered inoperable during the performance of SRs.	SR 4.8.2.2	SR 3.8.5.1	None	unique
3.8.5 L.3	Adds an additional Action that states "Declare affected required feature(s) inoperable." This offers an option to the current requirement to suspend CORE ALTERATIONS, movement of irradiated fuel suspending operations involving positive reactivity additions.	LCO 3.8.3.2 Action	LCO 3.8.5 Action A.1	None	unique

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.8.6 L.1	Extends the Action Completion time to verify Category C limits to once per 7 days and allows 31 days to restore Category A and B limits.	LCO 3.8.2.1 Table 4.8-2 Footnotes (1), (2)	LCO 3.8.6 Action A.2, A.3	VII	
3.8.6 L.2	Provides an allowance that "...Level correction is not required, however, when battery charging is < 2 amps when on float charge."	LCO 3.8.2.1 Table 4.8-2 Footnote (3b)	LCO 3.8.6 Table 3.8.6-1 Footnote (b)	IX	
3.8.6 L.3	Provides an allowance that "It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing."	LCO 3.8.2.1 Table 4.8-2	LCO 3.8.6 Table 3.8.6-1 Footnote (a)	IX	
3.8.6 L.4	Not Used	N/A	N/A	N/A	
3.8.6 L.5	Not Used	N/A	N/A	N/A	
3.8.7 None	None	N/A	N/A	N/A	
3.8.8 L.1	Not Used	N/A	N/A	N/A	
3.8.8 L.2	Adds an additional Action that states "Declare affected required feature(s) inoperable." This offers an option to the current requirement to suspend CORE ALTERATIONS, movement of irradiated fuel suspending operations involving positive reactivity additions.	LCO 3.8.3.2	LCO 3.8.8 Action A.1	None	unique
3.8.9 L.1	Not Used	N/A	N/A	N/A	
3.8.9 L.2	Not Used	N/A	N/A	N/A	
3.8.10L.1	Not Used	N/A	N/A	N/A	
3.8.10L.2	Not Used	N/A	N/A	N/A	

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Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.8.10L.3	Adds an additional Action that states "Declare affected required feature(s) inoperable." This offers an option to the current requirement to suspend CORE ALTERATIONS, movement of irradiated fuel suspending operations involving positive reactivity additions.	LCO 3.8.3.2	LCO 3.8.10 Action A.1	None	unique
3.9.1 None	None	N/A	N/A	N/A	
3.9.2 L.1	Eliminates the use of the term "not operating" in the Actions related to the SRMs functional condition.	LCO 3.9.2 Action a, b	LCO 3.9.2 Action A, B	None	unique
3.9.2 L.2	Eliminates the requirement to perform a CHANNEL FUNCTIONAL TEST on the SRMs since they do not provide control or alarm functions.	SR 4.9.2.b SR 4.9.2.c	LCO 3.9.2	IX	
3.9.3 L.1	Eliminates the requirement to use a manual actuation to test the containment purge valves and permits the flexibility to use an actual actuation to be credited for the Surveillance.	SR 4.9.9	SR 3.9.3.2	VI	
3.9.3 L.2	Relaxes the Surveillance Interval for demonstrating containment purge valve automatic isolation, going from within 72 hours prior to start of and once per 7 days during CORE ALTERATIONS to 18 months.	SR 4.9.9	SR 3.9.3.2	II	

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IV CTS Allowed Outage Time Extensions from 24 to 72 Hours
V CTS Action Requirements for Exiting LCOs are Changed
VI CTS Surveillance Acceptance Criteria Are Changed
VII Other CTS Allowed Outage Time Extensions
VIII Elimination of CTS Reporting Requirements
IX Relaxation of LCO Requirements

Discussion of Change	Description	CTS Section	ITS Section	Category	Characterization
3.9.3 L.3	Expands the type of closure device acceptable for piping penetrations (other than purge valve penetrations) which provide direct access from the containment atmosphere to outside atmosphere.	LCO 3.9.4.c.1	LCO 3.9.3 Action C.1	IX	
3.9.4 None	None	N/A	N/A	N/A	
3.9.5 None	None	N/A	N/A	N/A	
3.9.6 L.1	Eliminates the requirement to perform the Refueling Water Level Surveillance within 2 hours prior to the start of irradiated fuel assembly movement.	SR 4.9.10.1	SR 3.9.6.1	II	
3.9.7 L.1	Eliminates the requirement to perform the Refueling Water Level Surveillance within 2 hours prior to the start of irradiated fuel assembly movement.	SR 4.9.10.2	SR 3.9.7.1	II	
4.0 None	None	N/A	N/A	N/A	
5.0 L.1	Changes the responsibility for review of proposed tests and experiments which affect nuclear safety and are not addressed in the UFSAR or Technical Specifications from "Vice President Nuclear Production or his designee" to "Department Leader, Operations"	Section 6.5.2.5	Section 5.1.1	None	unique
5.0 LB.1	Changes the distance the dose is measured from, the source of radioactivity from 18 inches to 30 centimeters. This is consistent with the changes to 10CFR20	6.12.2	5.7.2	None	unique
5.0 LB.2	Not Used	N/A	N/A	N/A	N/A

Categories:

I CTS LCO Applicability Changes
II CTS Surveillance Frequency Changes
III CTS LCO Revised to Address Train Configurations
IV CTS Allowed Outage Time Extensions from 24 to 72 Hours
V CTS Action Requirements for Exiting LCOs are Changed
VI CTS Surveillance Acceptance Criteria Are Changed
VII Other CTS Allowed Outage Time Extensions
VIII Elimination of CTS Reporting Requirements
IX Relaxation of LCO Requirements

Table of PVNGS Relocated Specifications (R)

CTS Section No.	CTS LCO #	CTS LCO Title
3.1	3.1.2.1	Flow Paths - Shutdown
3.1	3.1.2.2	Flow Paths - Operating
3.1	3.1.2.3	Charging Pumps - Shutdown
3.1	3.1.2.4	Charging Pumps - Operating
3.1	3.1.2.5	Borated Water Sources - Shutdown
3.1	3.1.2.6	Borated Water Sources - Operating
3.3	3.3.1 Table 3.3-1	Supplementary Protection System (SPS)
3.3	3.3.3.1 Table 3.3-6	Fuel Pool Area Monitor RU-31
3.3	3.3.3.1 Table 3.3-6	New Fuel Area Monitor RU-19
3.3	3.3.3.1 Table 3.3-6	Main Steam Area Monitors (RU-139 A&B, RU-140 A&B)
3.3	3.3.3.1 Table 3.3-6	Post Accident Sampling System
3.3	3.3.3.2	Incore Detectors
3.3	3.3.3.3	Seismic Monitoring
3.3	3.3.3.4	Meteorological Instrumentation
3.3	3.3.3.6	Post Accident Monitoring Instrumentation
3.3	3.3.3.7	Loose-Parts Detection System
3.3	3.3.3.8	Explosive Gas Monitoring Instrumentation
3.4	3.4.3.2	Auxiliary Spray
3.4	3.4.6	RCS Chemistry
3.4	3.4.8.2	Pressurizer Heatup/Cooldown Limits
3.4	3.4.9	Structural Integrity
3.4	3.4.10	Reactor Coolant System Vents (Reactor Vessel Head Vents only)
3.6	3.6.4.3	Hydrogen Purge Cleanup System
3.7	3.7.2	Steam Generator Pressure/Temperature Limitation
3.7	3.7.9	Snubbers
3.7	3.7.10	Sealed Source Contamination

CTS Section No.	CTS LCO #	CTS LCO Title
3.8	3.8.1.3	Cathodic Protection
3.8	3.8.4.1	Containment Penetration Conductor Overcurrent Protection Devices
3.8	3.8.4.2	Motor Operated Valve Thermal Overload Protection and Bypass Devices
3.9	3.9.3	Decay Time
3.9	3.9.5	Communications
3.9	3.9.6	Refueling Machine
3.9	3.9.7	Crane Travel-Spent Fuel Storage Pool
3.9	3.9.12	Fuel Building Essential Ventilation
3.10	3.10.3	Special Test Exception - Reactor Coolant Loops
3.10	3.10.6	Special Test Exception - Safety Injection Tanks
3.10	3.10.7	Special Test Exception - Spent Fuel Pool Level
3.10	3.10.8	Special Test Exception - Safety Injection Tank
3.10	3.10.9	Special Test Exception - Shutdown Margin and Kn-1 CEDMS

Enclosure 3

Proposed License Conditions

Enclosure 3

Proposed License Conditions

APS requests that the Additional License Conditions 1 and 2 below be added to each Appendix D to the Palo Verde Nuclear Generating Station Facility Operating Licenses, No. NPF-41 for PVNGS Unit 1, No. NPF-51 for PVNGS Unit 2, and NPF-74 for Unit 3.

APS also requests that the Additional License Conditions currently in each Appendix D be deleted because they have been implemented as required and serve no further purpose. Marked-up Appendix D pages are attached.

Additional Condition 1

This amendment authorizes the relocation of certain technical specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the Table of PVNGS Relocated Details (LA) and table XX [the table of quality assurance-related relocations beginning on page 32 of the draft SE] in the Safety Evaluation enclosed with this amendment, as evaluated in the SE.

Implementation Date for Additional Condition 1

The relocation must be completed prior to or upon implementation of this amendment.

For the licensee-controlled documents that require updates to be submitted to the NRC in accordance with regulations or technical specifications, the next required updates after the implementation of this amendment shall contain the changes associated with the relocated technical specifications.

Enclosure 3

Proposed License Conditions (Continued)

Additional Condition 2

For surveillance requirements (SRs) that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to the date of implementation of this amendment.

Implementation Date for Additional Condition 2

This additional condition is effective upon implementation of this amendment.

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-41

Arizona Public Service Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
111	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated May 2, 1995, as supplemented by letter dated March 7, 1996, and evaluated in the staff's Safety Evaluation dated March 17, 1997.	60 days from the date of issuance.
112	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated December 27, 1996, as supplemented by letter dated March 18, 1997, and evaluated in the staff's Safety Evaluation dated March 26, 1997.	60 days from the date of issuance.

↑ delete

add new additional conditions 1 and 2

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-51

Arizona Public Service Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
103	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated May 2, 1995, as supplemented by letter dated March 7, 1996, and evaluated in the staff's Safety Evaluation dated March 17, 1997.	60 days from the date of issuance.
104	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated December 27, 1996, as supplemented by letter dated March 18, 1997, and evaluated in the staff's Safety Evaluation dated March 28, 1997.	60 days from the date of issuance.

↖ delete

add new Additional Conditions 1 and 2

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-74

Arizona Public Service Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
83	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated May 2, 1995, as supplemented by letter dated March 7, 1996, and evaluated in the staff's Safety Evaluation dated March 17, 1997.	60 days from the date of issuance.
84	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated December 27, 1996, as supplemented by letter dated March 18, 1997, and evaluated in the staff's Safety Evaluation dated March 26, 1997.	60 days from the date of issuance.

↖ delete

add new Additional Conditions 1 and 2

ENCLOSURE 6

Summary of ITS, ITS Bases, and ITS Justification Package Changes Included in This Submittal

Summary of ITS, ITS Bases, and ITS Justification Package Changes Included in This Submittal

ITS Justification Package Change:

Section 3.1, "Reactivity Control"

DOC LA.1 of specification 3.1.7, "Regulating CEA Insertion Limits" has been changed to indicate that this requirement is being relocated to the Core Operating Limits Report (COLR). The PVNGS ITS Table for LA changes is also being revised accordingly. No change to the ITS, NUREG markup or CTS markup are needed.

ITS Justification Package Changes:

Section 3.3, "Instrumentation"

- DOC L.2 of specification 3.3.5, "ESFAS Instrumentation" has been changed to more accurately reflect the change as marked up on the CTS markup and to be consistent with DOC L.5 of Specification 3.3.2, "ESFAS Instrumentation". No change to the ITS, NUREG markup or CTS markup is needed. No change to the ITS Tables are needed.

- The CTS markup to CTS Table 4.3-2, Table notation (3) for change L.6 of specification 3.3.5, "ESFAS Instrumentation" has been changed to be consistent with the requirements in the ITS. No change to the ITS, NUREG markup or ITS Tables are needed.

ITS Bases Change:

Section 3.5 "ECCS"

The references to minimum borated SIT water volumes that are expressed in cubic feet in the background section of the Bases for ITS section 3.5.2, "SITs Shutdown" have been removed. The SIT cubic feet requirements are contained in the Safety Analysis section of the Bases of ITS 3.5.2 and have been updated under 10 CFR 50.59. No change to the ITS, CTS markup, DOC's or the ITS Tables are needed.

Summary of ITS, ITS Bases, and ITS Justification Package Changes Included in This Submittal (continued)

ITS Bases Change:

Section 3.7, "Plant Systems"

The wording of the Bases for specification 3.7.7, "EW System" for isolation of EW components was revised in the LCO section of the Bases to be consistent with the wording in the SR 3.7.7.1 section of the Bases for ITS 3.7.7. No change to the ITS, CTS markup, DOC's or the ITS Tables are needed.

ITS Justification Package Changes:

Section 3.8, "Electrical 'Power Systems"

- NUREG Exception No. 19 to specification 3.8.1, "AC Sources-Operating" has been changed to provide additional justification for the interval of SR 3.8.1.5. No change to the ITS, ITS Bases, DOC's, NUREG markup, CTS markup or the ITS Tables are needed.
- The applicable NUREG Exceptions for the shutdown specifications in Section 3.8 (specification 3.8.2, "AC Sources-Shutdown", specification 3.8.5, "DC Sources-Shutdown", specification 3.8.8, "Inverters-Shutdown" and specification 3.8.10, "Distribution Systems-Shutdown") were revised to provide additional justification to add a note that states that suspension of positive reactivity additions is not applicable in MODES 1, 2, 3 and 4. No change to the ITS, ITS Bases, DOC's, NUREG markup, CTS markup or the ITS Tables are needed.
- The applicable DOC's for the shutdown specifications in Section 3.8 (specification 3.8.2, "AC Sources-Shutdown", specification 3.8.5, "DC Sources-Shutdown", specification 3.8.8, "Inverters-Shutdown" and specification 3.8.10, "Distribution Systems-Shutdown") were revised to provide additional justification to add a note that states that suspension of positive reactivity additions is not applicable in MODES 1, 2, 3 and 4. No change to the ITS, ITS Bases, NUREG markup, CTS markup or the ITS Tables are needed.

ITS Changes:

Chapter 5.0, "Administrative Control"

- The reference to the Testing Code "AG-1" in the Ventilation Filter Testing Program (VFTP) in ITS section 5.5.11 has been removed.

This code is not part of the PVNGS current licensing bases and is not appropriate for PVNGS. No change to the CTS markup, DOC's or the ITS Tables are needed.

- The first sentence in ITS 5.7.1, "High Radiation Area" has been revised to clarify that the requirements of the ITS are alternatives pursuant to 10CFR20.1601(c). This change is consistent with the current operating practices at PVNGS. No changes to the ITS Tables are needed.

ITS Justification Package Changes:

Changes to ITS Summary Tables

"LA" Table:

- For DOC 3.1.7 LA.1, the Destination Document should be changed to the COLR.

- On the last page of the "LA" Table, the Core Operating Limits Report (COLR) should be added to the list of documents along with the Methods of Control. The method of controls for the COLR are ITS Section 5.6 and 10 CFR 50.59.

"L" Table

For DOC 3.3.5 L.1, the category should be changed to category IX. The unique characterization should be deleted.

50-528 PALO VERDE 1 APSCO

APS COMMENTS ON THE DRAFT NRC SAFETY EVALUATION FOR PROPOSED CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS BOOK #2

REC'D W/LTR DTD 3/27/98...9804130185

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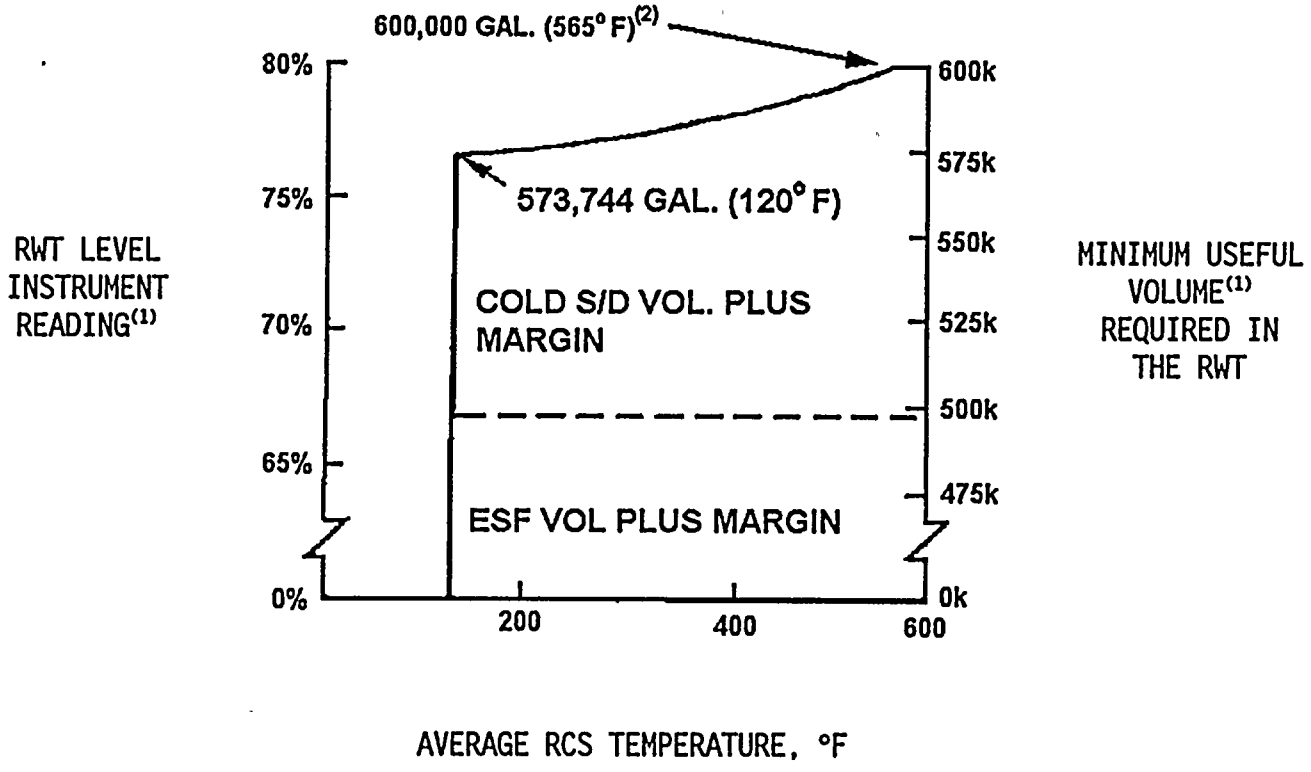
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	----------NOTE---------- Only required to be performed when ambient air temperature is < 60°F or > 120°F. ---------- Verify RWT borated water temperature is ≥ 60°F and ≤ 120°F.	24 hours
SR 3.5.5.2	Verify RWT borated water volume is ≥ minimum required RWT volume in Figure 3.5.5-1.	7 days
SR 3.5.5.3	Verify RWT boron concentration is ≥ 4000 ppm and ≤ 4400 ppm.	7 days



(1) The tank level and volume shown are the useful level and volume above that in the tank which is required for vortex considerations.

(2) When average RCS temperature is > 565°F, the minimum useful volume required in the RWT is 600,000 gallons.

FIGURE 3.5.5-1
Minimum Required RWT Volume

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.6 Trisodium Phosphate (TSP)

LCO 3.5.6 The TSP baskets shall contain ≥ 524 ft^3 of active TSP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. TSP not within limits.	A.1	Restore TSP to within limits.	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 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the TSP baskets contain ≥ 524 ft^3 of granular anhydrous trisodium phosphate.	18 months
SR 3.5.6.2	Verify that a sample from the TSP baskets provides adequate pH adjustment of borated water.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment inoperable.	A.1	Restore containment to OPERABLE status.	1 hour
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

----------NOTES----------

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ----------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock. AND	1 hour
	A.2 Lock the OPERABLE door closed in the affected air lock. AND	24 hours
	A.3 --------NOTE-------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	----------NOTES---------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ----------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1	Verify an OPERABLE door is closed in the affected air lock.	1 hour
	AND		
	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	--------NOTE-------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----------	
		Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

----------NOTES----------

1. Penetration flow paths except for 42 inch purge valve penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable except for purge valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTE-------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with two containment isolation valves inoperable except for purge valve leakage not within limit.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND C.2 --------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	4 hours Once per 31 days
D. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve with resilient seals, or blind flange. AND	24 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	--------NOTE-------- Isolation devices in high radiation areas may be verified by use of administrative means. ----------	
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND		
	D.3	Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each 42 inch purge valve is sealed closed except for one purge valve in a penetration flow path while in Condition D of this LCO.	31 days
SR 3.6.3.2	Verify each 8 inch purge valve is closed except when the 8 inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	31 days
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.4	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for containment purge valves with resilient seals.	184 days AND Within 92 days after opening the valve
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.3 psig and ≤ +2.5 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 117°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when RCS pressure is ≥ 385 psia

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One containment spray train inoperable.	A.1	Restore containment spray train to OPERABLE status.	72 hours
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 4 with RCS pressure < 385 psia.	84 hours
C. Two containment spray trains inoperable.	C.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.6.1	Verify each containment spray 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
SR 3.6.6.2	Verify the containment spray piping is full of water to the 113 ft level in the containment spray header.	31 days
SR 3.6.6.3	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.4	Verify each automatic containment spray 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
SR 3.6.6.5	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.6	Verify each spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shared among the three units shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------

All three PVNGS Units (Units 1, 2, and 3) shall simultaneously comply with the REQUIRED ACTION(s) when the shared portion of the hydrogen recombiner(s) is the cause of a CONDITION.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombiner inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Restore hydrogen recombiner to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained.	1 hour AND Every 12 hours thereafter
	AND	
	B.2 Restore one hydrogen recombiner to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	6 months
SR 3.6.7.2	Perform a functional test for each hydrogen recombiner.	6 months
SR 3.6.7.3	Perform a CHANNEL CALIBRATION to include a System Functional Test for each hydrogen recombiner.	12 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1	Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours
	AND		
	A.2	Reduce the variable overpower trip–high setpoint in accordance with Table 3.7.1-1.	12 hours
B. Required Action and associated Completion Time not met. OR One or more required steam generators with less than six MSSVs OPERABLE.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	----------NOTE---------- Not required to be performed prior to entry into MODE 3. ---------- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)	MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)
10	100.0	111.0
9	98.2	108.0
8	87.3	97.1
7	76.4	86.2
6	65.5	75.3

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 3%)
Steam Generator #1	Steam Generator #2	
SGE PSV 572	SGE PSV 554	1250
SGE PSV 579	SGE PSV 561	1250
SGE PSV 573	SGE PSV 555	1290
SGE PSV 578	SGE PSV 560	1290
SGE PSV 574	SGE PSV 556	1315
SGE PSV 575	SGE PSV 557	1315
SGE PSV 576	SGE PSV 558	1315
SGE PSV 577	SGE PSV 559	1315
SGE PSV 691	SGE PSV 694	1315
SGE PSV 692	SGE PSV 695	1315

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2, 3, and 4 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1	Restore MSIV to OPERABLE status.	4 hours
B. Required Action and Associated Completion Time of Condition A not met.	B.1	Be in MODE 2.	6 hours
C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV. ---------- One or more MSIVs inoperable in MODE 2, 3, or 4.	C.1 AND C.2	Close MSIV. Verify MSIV is closed.	4 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 AND D.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	----------NOTE---------- Not required to be performed prior to entry into MODE 3. ---------- Verify closure time of each MSIV is ≤ 4.6 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs)

LCO 3.7.3 Four economizer MFIVs and four downcomer MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 except when MFIV is closed or isolated by a closed power operated valve.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more MFIVs inoperable.	A.1	Close or isolate inoperable MFIV.	72 hours
	AND		
	A.2	Verify inoperable MFIV is closed or isolated.	Once per 7 days
B. Two valves in the same flow path inoperable.	B.1	Isolate affected flow path.	8 hours
	AND		
	B.2	Verify inoperable MFIV is closed or isolated.	Once per 7 days

(continued

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV is ≤ 9.6 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

LCO 3.7.4 One ADV line per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ADV line inoperable.	A.1	--------NOTE-------- LCO 3.0.4 is not applicable. ---------- Restore ADV line to OPERABLE status.	72 hours
B. Two required ADV lines inoperable.	B.1	Restore one ADV line to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	18 months

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

---NOTE---
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1 Restore AFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR Two AFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours
D. Three AFW trains inoperable in MODE 1, 2, or 3.	D.1 --------NOTE-------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	 Immediately
E. Required AFW train inoperable in MODE 4.	E.1 --------NOTE-------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 72 hours after reaching 532°F in the RCS. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Test Program
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 72 hours after reaching 532°F in the RCS. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 72 hours after reaching 532°F in the RCS. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify the proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST level shall be ≥ 29.5 ft.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. CST level not within limit.	A.1	Verify OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND		
	A.2	Restore CST level to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify CST level is ≥ 29.5 ft.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Essential Cooling Water (EW) System

LCO 3.7.7 Two EW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EW train inoperable.	A.1	--------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4" for shutdown cooling made inoperable by EW. ----------	
		Restore EW train to OPERABLE status.	72 hours
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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	----------NOTE---------- Isolation of EW flow to individual components does not render the EW System inoperable. ---------- Verify each EW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each EW 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
SR 3.7.7.3	Verify each EW pump starts automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.8 Essential Spray Pond System (ESPS)

LCO 3.7.8 Two ESPS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ESPS train inoperable.	A.1	--------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources – Operating," for emergency diesel generator made inoperable by ESPS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4," for shutdown cooling made inoperable by ESPS. ---------- Restore ESPS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	----------NOTE---------- Isolation of ESPS flow to individual components does not render ESPS inoperable. ---------- Verify each ESPS manual and power operated valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each ESPS pump starts automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. UHS inoperable.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify the usable water depth of each essential spray pond is ≥ 12 feet.	24 hours
SR 3.7.9.2	Verify water temperature of each essential spray pond is ≤ 89°F.	24 hours

3.7 PLANT SYSTEMS

3.7.10 Essential Chilled Water (EC) System

LCO 3.7.10 Two EC trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EC train inoperable.	A.1	Restore EC train 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each EC System 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
SR 3.7.10.2	Verify the proper actuation of each EC System component on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.11 Control Room Essential Filtration System (CREFS)

LCO 3.7.11 Two CREFS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies,

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREFS train inoperable.	A.1	Restore CREFS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODES 5 and 6.	C.1	Place OPERABLE CREFS train in operation.	Immediately
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 OR D.2	Place OPERABLE CREFS Train in operation. Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREFS trains inoperable in MODES 5 and 6, or during movement of irradiated fuel assemblies.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
F. Two CREFS trains inoperable in MODE 1, 2, 3, or 4.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each CREFS train for ≥ 15 minutes.	31 days
SR 3.7.11.2	Perform required CREFS filter testing in accordance with Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.11.3	Verify each CREFS train actuates on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.11.4	Verify one CREFS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the adjacent area during operation at a ventilation flow rate of ≤ 1000 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.12 Two CREATCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies,

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREATCS train inoperable.	A.1	Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6.	C.1	Place OPERABLE CREATCS train in operation.	Immediately
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 OR D.2	Place OPERABLE CREATCS train in operation Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREATCS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
F. Two CREATCS trains inoperable in MODE 1, 2, 3, or 4.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify each CREATCS train has the capability to remove the assumed heat load.	18 months

3.7 PLANT SYSTEMS

3.7.13 Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.13 Two ESF PREACS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ESF PREACS train inoperable.	A.1	Restore ESF PREACS train to OPERABLE status.	7 days
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each ESF PREACS train for ≥ 15 minutes	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.2	Perform required ESF PREACS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each ESF PREACS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.4	Verify one ESF PREACS train can maintain a measurable negative pressure relative to atmospheric pressure during operation at a flowrate of 6000 cfm $\pm$10%.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.14 Fuel Storage Pool Water Level

LCO 3.7.14 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Boron Concentration

LCO 3.7.15 The fuel storage pool boron concentration shall be ≥ 2150 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.16 Secondary Specific Activity

LCO 3.7.16 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the specific activity of the secondary coolant is within limit.	31 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment and burnup of each fuel assembly stored in each of the three regions of the fuel storage pool shall be within the acceptable burnup domain for each region as shown in Figure 3.7.17-1 and described in Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

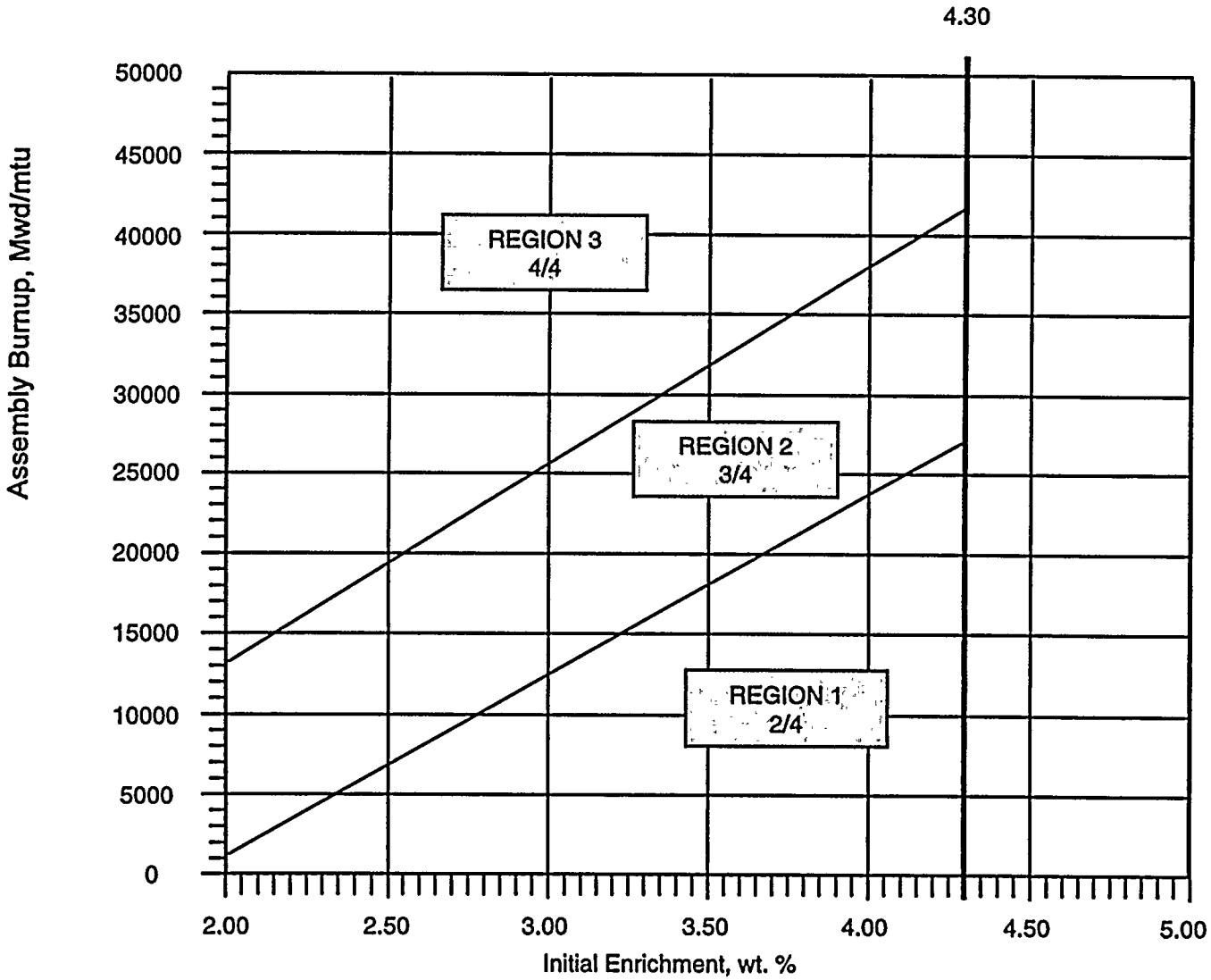
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly into an appropriate region.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 and Specification 4.3.1.1.	Prior to storing the fuel assembly in the fuel storage pool.

FIGURE 3.7.17-1
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT



3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System; and

c. Automatic load sequencers for Train A and Train B.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for the OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.4	Restore DG to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating," when Condition D is entered with no AC power source to a train. ---------- D.1 Restore required offsite circuits to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours
F. One automatic load sequencer inoperable.	F.1 Restore automatic load sequencer to OPERABLE status. AND F.2 Declare required feature(s) supported by the inoperable sequencer inoperable when its redundant required feature(s) is inoperable.	24 hours 4 hours from discovery of Condition F concurrent with inoperability of redundant required feature(s)

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Electrical Distribution System input voltage less than limits.	G.1 Block one train of fast bus transfer AND	1 hour
	G.2.1 Block the opposite train of fast bus transfer OR	2 hours
	G.2.2.1 Start, load and separate the opposite train DG from offsite power AND	2 hours
	G.2.2.2 Restore Electrical Distribution System input voltage to within limits	72 hours
H. Required Action and Associated Completion Time of Condition A, B, C, D, E, F, or G not met.	H.1 Be in MODE 3. AND	6 hours
	H.2 Be in MODE 5.	36 hours
I. Three or more required AC sources inoperable.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	----------NOTES---------- 1. Performance of SR 3.8.1.7 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. ---------- Verify each DG starts from standby condition and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ---------- Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 4950 kW and ≤ 5500 kW.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 550 gal of fuel oil (minimum level of 2.75 feet).	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	92 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from the storage tank to the day tank.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	----------NOTE---------- All DG starts may be preceded by an engine prelube period followed by a warmup period prior to loading. ---------- Verify each DG starts from standby condition and achieves a. In ≤ 10 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.7 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8	----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. ---------- Verify manual transfer of AC power sources from the normal offsite circuit to each alternate offsite circuit.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 64.5 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.	18 months
SR 3.8.1.10	----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. ---------- Verify each DG does not trip, and voltage is maintained ≤ 6200 V during and following a load rejection of ≥ 4950 kW and ≤ 5500 kW.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	----------NOTE---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Momentary voltage and frequency transients induced by load changes do not invalidate this test. ---------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through automatic load sequencer, 3. maintains steady state voltage ≥ 3740 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 59.7 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal (without a loss of offsite power) each DG auto-starts and: a. In ≤ 10 seconds, achieves voltage ≥ 3740 V and frequency ≥ 58.8 Hz; b. Achieves steady state voltage ≥ 3740 and ≤ 4580 V and frequency ≥ 59.7 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes on standby (running unloaded); d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized (auto-connected through the automatic load sequencer) from the offsite power system.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. ---------- Verify each DG automatic trip is bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal except: a. Engine overspeed; b. Generator differential current; c. Engine low lube oil pressure; and d. Manual emergency stop trip.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	----------NOTES---------- 1. Momentary transients outside the load range do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. 3. All DG starts may be preceded by an engine prelube period followed by a warmup period prior to loading. 4. DG loading may include gradual loading as recommended by the manufacturer. ---------- Verify each DG operates for ≥ 24 hours: a. For ≥ 22 hours loaded ≥ 4950 kW and ≤ 5500 kW; and b. For the remaining hours (≥ 2) of the test loaded ≥ 5775 kW and ≤ 6050 kW.	18 months

Continued

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.15 ----------NOTES---------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG, loaded ≥ 4950 kW and ≤ 5500 kW, has operated ≥ 2 hours or until temperatures have stabilized. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts and achieves a. In ≤ 10 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.7 Hz and ≤ 61.2 Hz.	18 months
SR 3.8.1.16 ----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.17	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation; and b. Automatically energizing the emergency load from offsite power.	18 months
SR 3.8.1.18	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify interval between each sequenced load block is within ± 1 second of design interval for each automatic load sequencer.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage ≥ 3740 V and ≤ 4580 V, 4. achieves steady state frequency ≥ 59.7 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. ---------- Verify, when started simultaneously, each DG achieves a. In ≤ 10 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.7 Hz and ≤ 61.2 Hz.	10 years

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources – Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems Shutdown"; and

b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

----------NOTE----------
In MODES 1, 2, 3, and 4, Required Actions A.2.3 and B.3 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --- A.1 Declare affected required feature(s) with no offsite power available inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	A.2.3	Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND		
	A.2.4	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.3	Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.6, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.13 through SR 3.8.1.16, and SR 3.8.1.18. ---------- For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.8, SR 3.8.1.12, SR 3.8.1.17, SR 3.8.1.19 and SR 3.8.1.20, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel level < 80% indicated fuel level and ≥ 71% indicated fuel level in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory < 2.5 inches and > 1 inch visible in the sightglass.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more DGs with stored fuel oil total particulates not within limits.	C.1 Restore fuel oil total particulates to within limits.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with a required starting air receiver pressure < 230 psig and ≥ 185 psig.	E.1 Restore starting air receiver pressure to ≥ 230 psig.	48 hours
F. Required Action and associated Completion Time not met. OR One or more DGs with diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 80% indicated fuel level.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 2.5 inches visible in the sightglass.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG starting air receiver pressure is ≥ 230 psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DC electrical power subsystem (exclusive of the battery charger) inoperable.	A.1	Restore DC electrical power subsystem to OPERABLE status.	2 hours
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
C. Required DC electrical power subsystem battery charger inoperable.	C.1	Verify battery cell parameters meet Table 3.8.6-1 category A limits	1 hour AND Once per 8 hours thereafter
	AND		
	C.2	Restore DC electrical power subsystem battery charger to OPERABLE status	24 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1	Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 129 V on float charge (low specific gravity cells) or ≥ 131 V on float charge (AT&T).	7 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is ≤ 150E-6 ohms for inter-cell connections. ≤ 150E-6 ohms for inter-rack connections. ≤ 150E-6 ohms for inter-tier connections. and ≤ 150E-6 ohms for terminal connections.	92 days
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	18 months
SR 3.8.4.4	Remove visible terminal corrosion and verify battery cell to cell and terminal connections are clean, and are coated with anti-corrosion material.	18 months
SR 3.8.4.5	Verify battery connection resistance is ≤ 150E-6 ohms for inter-cell connections. ≤ 150E-6 ohms for inter-rack connections. ≤ 150E-6 ohms for inter-tier connections. and ≤ 150E-6 ohms for terminal connections.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.6	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4 on the charger credited for OPERABILITY. ---------- Verify each battery charger supplies ≥ 400 amps for Batteries A and B and ≥ 300 amps for Batteries C and D at ≥ 125 V for ≥ 8 hours.	18 months
SR 3.8.4.7	----------NOTES---------- 1. The battery performance discharge test or the modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----------	
	Verify battery capacity is ≥ 80% (low specific gravity cells) or ≥ 90% (AT&T) of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months AND 12 months when battery shows degradation or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources – Shutdown

LCO 3.8.5 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

----------NOTE----------
In MODES 1, 2, 3, and 4, Required Action A.2.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems (exclusive of the battery charger) inoperable.	A.1 Declare affected required feature(s) inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND A.2.2 Suspend movement of irradiated fuel assemblies. AND A.2.3 Initiate action to suspend operations involving positive reactivity additions. AND	Immediately Immediately Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately
B. Required DC electrical power subsystem battery charger inoperable.	B.1 Verify battery cell parameters meet Table 3.8.6-1 category A limits. AND B.2 Restore DC electrical power subsystem battery charger to OPERABLE status.	1 hour AND Once per 8 hours thereafter 24 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. ---------- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Train A and Train B batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1 Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits. AND	1 hour
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits. AND	24 hours AND Once per 7 days thereafter
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 60°F. OR One or more batteries with one or more battery cell parameters not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR	3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 7 days after battery discharge < 105 V AND Once within 7 days after battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 60°F.	92 days

Table 3.8.6-1 (page 1 of 2) (Low Specific Gravity Cells)
Battery Surveillance Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark(a)	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark(a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V(d)	> 2.07 V
Specific Gravity(b)(c)	≥ 1.200	≥ 1.195 AND Average of all connected cells ≥ 1.205	Not more than 0.020 below average connected cells AND Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

(d) Corrected for average electrolyte temperature.

Table 3.8.6-1 (page 2 of 2) (AT&T)
Battery Surveillance Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark(a)	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark(a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.18 V	≥ 2.18 V(d)	> 2.14 V
Specific Gravity(b)(c)	≥ 1.290	≥ 1.280 AND Average of all connected cells ≥ 1.290	Not more than 0.020 below average connected cells AND Average of all connected cells ≥ 1.280

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

(d) Corrected for average electrolyte temperature.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 The required Train A and Train B inverters shall be OPERABLE.

----------NOTE----------
One inverter may be disconnected from its associated DC bus for ≤ 24 hours to perform an equalizing charge on its associated battery, provided:

a. The associated AC vital instrument bus is energized from its Class 1E constant voltage source regulator; and

b. All other AC vital instrument buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 ---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital instrument bus de-energized. ---------- Restore inverter to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital instrument buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters – Shutdown

LCO 3.8.8 Required inverter(s) shall be OPERABLE to support the onsite Class 1E AC vital instrument bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
In MODES 1, 2, 3, and 4, Required Action A.2.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage, frequency, and alignments to required AC vital instrument buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems – Operating

LCO 3.8.9 Train A and Train B AC, DC, and AC vital instrument bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystem inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
B. One AC vital instrument bus electrical power distribution subsystem inoperable.	B.1 Restore AC vital instrument bus electrical power distribution subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One DC electrical power distribution subsystems inoperable.	C.1	Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time not met.	D.1 AND	Be in MODE 3.	6 hours
	D.2	Be in MODE 5.	36 hours
E. Two or more inoperable distribution subsystems that result in a loss of safety function.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital instrument bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems – Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC vital instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

----------NOTE----------
In MODES 1, 2, 3, and 4, Required Action A.2.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital instrument bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. OR	Immediately
	A.2.1 Suspend CORE ALTERATIONS. AND	Immediately
	A.2.2 Suspend movement of irradiated fuel assemblies. AND	Immediately
	A.2.3 Initiate action to suspend operations involving positive reactivity additions. AND	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, DC, and AC vital instrument bus electrical power distribution subsystems to OPERABLE status. AND A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital instrument bus electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two startup range monitors (SRMs) shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required SRM inoperable.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
B. Two required SRMs inoperable.	B.1	Initiate action to restore one SRM to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.9 REFUELING OPERATIONS

3.9.4 Shutdown Cooling (SDC) and Coolant Circulation-High Water Level

LCO 3.9.4 One SDC loop shall be OPERABLE and in operation.

---NOTE---
The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDC loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy SDC loop requirements.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC loop is operable and in operation.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Shutdown Cooling (SDC) and Coolant Circulation–Low Water Level

LCO 3.9.5 Two SDC loops shall be OPERABLE, and one SDC loop shall be in operation.

---NOTE---
The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SDC loop inoperable.	A.1	Initiate action to restore SDC loop to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No SDC loop OPERABLE or in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one SDC loop to OPERABLE status and to operation.	Immediately
	AND		
	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify required SDC loops are OPERABLE and one SDC loop is in operation.	12 hours
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Water Level-Fuel Assemblies

LCO 3.9.6 Refueling water level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling water level not within limit.	A.1 Suspend movement of fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.7 Refueling Water Level-CEAs

LCO 3.9.7 Refueling water level shall be maintained ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel.

APPLICABILITY: During movement of CEAs within the reactor vessel, when the fuel assemblies seated within the reactor vessel are irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling water level not within limit.	A.1 Suspend movement of CEAs within the reactor vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling water level is ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel.	24 hours

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area. The site is comprised of approximately 4,050 acres. Site elevations range from 890 feet above mean sea level at the southern boundary to 1,030 feet above mean sea level at the northern boundary. The minimum distance from a containment building to the exclusion area boundary is 871 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy 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. Other cladding material may be used with an approved exemption.

4.2.2 Control Element Assemblies

The reactor core shall contain 76 full length and 13 part length control element assemblies (CEAs). The control material shall be boron carbide with Inconel Alloy 625 used as a wear absorber over a portion of the part length control element assemblies as approved by the NRC.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.30 weight percent;

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 UFSAR;

c. A nominal 9.5 inch center-to-center distance between adjacent storage cell locations.

d. Region 1: Fuel shall be stored in a checkerboard (two-out-of-four) storage pattern. Fuel that qualifies to be stored in Regions 1, 2, or 3 in accordance with Figure 3.7.17-1, may be stored in Region 1.

e. Region 2: Fuel shall be stored in a three-out-of-four storage pattern. Fuel that qualifies to be stored in Regions 2 or 3, in accordance with Figure 3.7.17-1, may be stored in Region 2.

f. Region 3: Fuel shall be stored in a four-out-of-four storage pattern. Only fuel that qualifies to be stored in Region 3, in accordance with Figure 3.7.17-1, shall be stored in Region 3.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.30 weight percent;

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 UFSAR;

(continued)

4.0 DESIGN FEATURES (continued)

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 UFSAR; and

d. A nominal 17 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Department Leader, Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Department Leader, Operations or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The Control Room Supervisor (CRS) shall be responsible for the control room command function. During any absence of the CRS 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 CRS 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.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

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.

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 shall be documented in the UFSAR;

b. The Vice President, Nuclear Production 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;

c. The Senior Vice President, Nuclear 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

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.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f 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.

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.

d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the working hour guidelines shall be authorized in advance by personnel at the Director level or designees, in accordance with approved administrative procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by these authorized individuals or designees to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

e. The Operations Department Leader or Operations Supervisor shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975 and ANSI/ANS 3.1-1978, except the Director, Site Radiation Protection shall meet or exceed the qualification of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

5.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operation (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

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;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

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 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

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;

b. Shall become effective after the approval of the Director, Site Chemistry; and

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

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

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.

5.5.2 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 portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sampling return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents 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.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

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:

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;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

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;

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;

e. Determination of cumulative and projected 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;

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;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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;

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

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 mrems/yr to any organ;

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;

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

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.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

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 Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

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 0, October 1971.

5.5.8 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. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel 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

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the Inservice Inspection of steam generator tubes to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

5.5.9.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.9-1.

5.5.9.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 5.5.9.4a.8.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2 Steam Generator Tube Sample Selection and Inspection (continued)

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

----------NOTE----------
In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

5.5.9.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection.* If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.9-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

5.5.9.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

(continued)

TABLE 5.5.9-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 5.5.9-2
STEAM GENERATORS TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S = 3\frac{N}{n}$ % Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

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:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

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.

5.5.11 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 1.52, Revision 2, and in accordance with Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 at the system flowrate specified below ± 10%.

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 ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, at the system flowrate specified as follows ± 10%:

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREFS	28,600 CFM
ESF PREACS	6,000 CFM

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 Guide 1.52, Revision 2, and ANSI N510-1980 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 at a temperature of 80°C ± 0.5°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤ 1.0%	70%
ESF PREACS	≤ 1.0%	70%

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. For each of the ESF systems, demonstrate 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 ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREFS	8.4 inches water gauge	28,600 CFM
ESF PREACS	8.4 inches water gauge	6,000 CFM

e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage
ESF PREACS	> 19 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control 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 unprotected outdoor 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".

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

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);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank 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

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste 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 the limits of 10 CFR Part 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.

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.

5.5.13 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 as referenced in the UFSAR. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use 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. Water and sediment are within the limits of ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

A change in the TS incorporated in the license; or

A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b 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).

5.5.15 Safety Functions Determination Program (SFDP)

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 limitations and remedial or compensatory actions may be identified to 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, 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 system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Functions Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to 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.

5.5.16 Containment Leakage Rate Testing Program

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. 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:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 14.5 psig.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

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.

(continued)

5.6 Reporting Requirements (continued)

5.6.2 Annual Radiological Environmental Operating Report (continued)

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. The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. 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.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste system, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted 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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the shutdown cooling system suction line relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Length CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

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:

1. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

7. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

8. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

9. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

10. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

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.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "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.

(continued)

5.6 Reporting Requirements (continued)

5.6.7 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.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of tubes plugged.

Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In addition to the provisions of 10 CFR 20.1601, the following controls provide an alternate method for controlling access to high radiation areas as provided by paragraph 20.1601(c) of 10 CFR part 20. High radiation areas, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technicians) or personnel continuously escorted by such individuals may be exempt from the REP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Section Leader or designated alternate in the REP.

(continued)

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Manager on duty or Radiation Protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the REP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose in excess of 1000 mrem (measurement made at 30 cm from source of radioactivity), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

Enclosure 5

Improved Technical Specifications (ITS) Bases

PVNGS

Palo Verde Nuclear Generating Station
Units 1, 2, and 3

Improved Technical Specifications

ITS Bases

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure 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 (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that 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.

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.

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 the 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.

(continued)

BASES

BACKGROUND (continued)

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

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 a 95% probability at a 95% confidence level (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 RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, flow rate 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.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Pressurizer Pressure—High trip;

b. Pressurizer Pressure—Low trip;

c. Variable Overpower—High trip;

d. Steam Generator Pressure—Low trip;

e. Local Power Density—High trip;

f. DNBR—Low trip;

g. Steam Generator Level—Low trip;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

h. Log Power Level–High trip;

i. Reactor Coolant Flow–Low trip; and

j. Steam Generator Safety Valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation used in the protection system design as a measure of the core power is proportional to core power.

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to 1.3, based on a statistical combination of CE-1 CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained. Maintaining the dynamically adjusted peak LHR to $\leq$ 21 kW/ft ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply 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

(continued)

BASES

APPLICABILITY (continued)

function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

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 violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 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 where this SL is not applicable and reduces the probability of fuel damage.

2.2.3

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

2.2.4

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1.1 or SL 2.1.1.2 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.5 (continued)

report shall also be provided to the senior management of the nuclear plant, and the utility Vice President, Nuclear Production.

2.2.6

If SL 2.1.1.1 or SL 2.1.1.2 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.
2. UFSAR, Sections 6 and 15.
3. 10 CFR 50.72.
4. 10 CFR 50.73.

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, continued RCS integrity 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 Boundary (RCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, according to GDC 28 (Ref. 1), "Reactivity Limits," reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept 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).

Overpressurization of the RCS could result in a breach of the RCPB. If this 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 Pressure–High trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of 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 Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure–High Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure–High 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 the following:

a. Steam Bypass Control System;

b. Pressurizer Level Control System; or

c. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS under the ASME Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is established at 2750 psia.

(continued)

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

The following SL violation responses are applicable to the RCS pressure SLs.

2.2.2.1

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.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater that the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

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 potentially 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. This action does not require reducing MODES, since this would require reducing temperature, which would

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.2.2 (continued)

compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 6).

2.2.4

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and to assess the condition of the unit before reporting to the senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 7). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President, Nuclear Production.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

(continued)

BASES

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.

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

4. 10 CFR 100.

5. UFSAR, Section 7.

6. 10 CFR 50.72.

7. 10 CFR 50.73.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.7 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

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

(continued)

BASES

LCO 3.0.2 (continued)

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.

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. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which 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.

(continued)

BASES

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

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.

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.

(continued)

BASES

LCO 3.0.3 (continued)

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.

b. A Condition exists for which the Required Actions have now been performed.

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 Specification 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 5 is the next 35 hours, because the total time for reaching MODE 5 is not reduced from the allowable limit of 37 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.

(continued)

BASES

LCO 3.0.3 (continued)

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. 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 3.7.14, "Fuel Storage Pool Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 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 of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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 precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

(continued)

BASES

LCO 3.0.4 (continued)

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. 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 including any non-LCO driven shutdowns such as entry into a refueling outage.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow 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. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4 . The requirements of LCO 3.0.4 do not apply in MODES 5 and 6 or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." . This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.

(continued)

BASES

LCO 3.0.4 (continued)

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, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated 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

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.

(continued)

BASES

LCO 3.0.5 (continued)

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 support systems that have an 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.

(continued)

BASES

LCO 3.0.6 (continued)

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 5.5.15, "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.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.0.6-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.0.6-2)

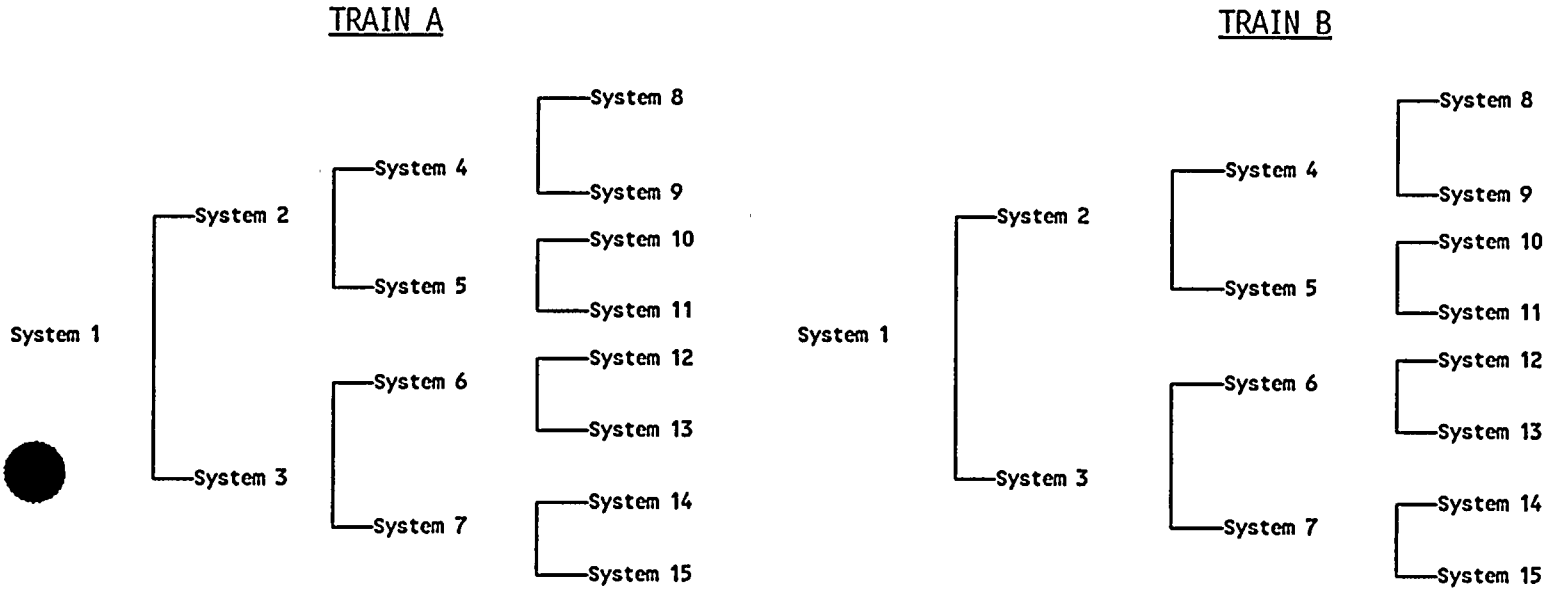
c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

(continued)

BASES

LCO 3.0.6 (continued) If this 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.

EXAMPLES



EXAMPLE B3.0.6-1
If System 2, of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

EXAMPLE B3.0.6-2
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 which is in turn supported by System 5.

EXAMPLE B3.0.6-3
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.

For the examples above, support systems are to the left of the supported systems (i.e., System 1 supports System 2 and System 3).

(continued)

BASES

LCO 3.0.7

Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, Special Test Exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

(continued)

BASES

LCO 3.0.7 (continued)

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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.

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 to be not 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 Special Test Exception (STE) are only applicable when the STE 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.

(continued)

BASES

SR 3.0.1 (continued)

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.

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 > 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.

b. High Pressure Safety Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

(continued)

BASES

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% 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).

The 25% 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 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Containment Leak Rate Testing Program.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% 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% 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.

(continued)

BASES

SR 3.0.2 (continued)

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 less, 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. Reference Bases Section 3.0.2 for discussion and applicability of Frequency and 25% extension.

This delay period provides an 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.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

(continued)

BASES

SR 3.0.3 (continued)

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.

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.

(continued)

BASES

SR 3.0.4 (continued)

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 instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 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 SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

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 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.

(continued)

BASES

SR 3.0.4 (continued)

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, Mode 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3 or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3 or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Open

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). 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 of all full length control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn with Reactor Trip Breakers open. This reactivity worth is credited in establishing the required SDM.

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 CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the CEA 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 CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for SDM 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 limit AOOs, and $\leq$ 280 cal/gm energy deposition for the CEA ejection accident).

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 initial RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine 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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3, 4, and 5 must also protect against:

a. Inadvertent boron dilution;

b. Startup of an inactive reactor coolant pump (RCP); and

c. CEA ejection.

Each of these 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.

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. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM.

The SDM satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

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 Criterion," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEAs) and through the soluble boron concentration.

APPLICABILITY

In MODES 3, 4 and 5 with the Reactor Trip Breakers Open or the CEA drive system not capable of CEA withdrawal, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. In MODES 3, 4 and 5 with the Reactor Trip Breakers Closed, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

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

(continued)

BASES (continued)

ACTIONS

A.1 (continued)

refueling water tank. The operator should borate with the best source available for the plant conditions.

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 will exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 26 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes with a 4000 ppm source.

If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 26 gpm and 4000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation. When taking credit for the Xenon concentration in the reactivity balance calculation, the frequency may have to be administratively controlled to ensure that SDM does not go below the limit due to Xenon decay.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 15.2.
3. UFSAR, Section 15.4.
4. 10 CFR 100.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). 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, SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full length control element assemblies (CEAs), assuming the single CEA 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 CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding the acceptable fuel design limits, assuming that the CEA 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 CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth CEA stuck out following a reactor trip. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for SDM requirements are that the 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, fuel centerline temperature limits for AOOs, and $\leq$ 280 cal/gm energy deposition for the CEA 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 initial RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine 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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3, 4, and 5 must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled CEA withdrawal from a subcritical condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. CEA ejection.

Each of these 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.

The withdrawal of CEAs from subcritical conditions 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 withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of the CEAs must be accounted for in the determination of SDM.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 350°F, the inadvertent deboration event becomes limiting with respect to the applicable SHUTDOWN MARGIN requirements. Below 350°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with the reactor trip breakers closed and the CEA drive system capable of CEA withdrawal, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. The K_{N-1} requirement ensures that a CEA ejection event while shutdown will not result in criticality. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

The requirement prohibiting criticality due to shutdown group CEA movement is associated with the assumptions used in the analysis of uncontrolled CEA withdrawal from subcritical conditions. Due to the high differential reactivity worth of the shutdown CEA groups, the analysis assumes that the initial shutdown reactivity is such that the reactor will remain subcritical in the event of unexpected or uncontrolled shutdown group withdrawal.

The SDM satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the reactivity control requirements 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 Criterion," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM, K_{N-1}, and criticality due to Shutdown CEA withdrawal are core physics design conditions that can be ensured through CEA positioning (regulating and shutdown CEAs) and through the soluble boron concentration.

APPLICABILITY

In MODES 3, 4, and 5 with the Reactor Trip Breakers Closed and the CEA Drive System is capable of CEA withdrawal, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. MODES 3, 4 and 5 with the Reactor Trip Breakers Open, SDM is addressed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Open." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

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 and/or vary CEA position. 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

(continued)

BASES

ACTIONS

A.1 (continued)

concentrated solution, such as that normally found in the refueling water tank. The operator should borate with the best source available for the plant conditions.

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 will exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 26 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes with a 4000 ppm source. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 26 gpm and 4000 ppm represent typical values and are provided for the purpose of offering a specific example.

B.1 and B.2

If the K_{N-1} requirements are not met or reactor criticality is achievable by Shutdown Group CEA movement, boration must be initiated promptly and CEA position varied to restore K_{N-1} within limit or to ensure criticality due to Shutdown Group CEA movement is not achievable. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components and vary CEA position. It is assumed that boration will be continued and CEA position varied to return K_{N-1} to within limit or prevent reactor criticality due to Shutdown Group CEA movement. CEA movement is only required if the specific limit exceeded can be improved by taking this action.

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 refueling water tank. The operator should borate with the best source available for the plant conditions.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 will exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 26 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes with a 4000 ppm source. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 26 gpm and 4000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1, 3.1.2.2 and 3.1.2.3

SDM, K_{N-1}, and criticality not being achievable with Shutdown Group CEA withdrawal are verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

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 that of the RCS.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1, 3.1.2.2 and 3.1.2.3 (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and it allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation. When taking credit for xenon concentration in the reactivity balance calculation, the frequency may have to be administratively controlled to ensure that SDM does not go below the limit due to Xenon decay.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 15.1.
3. UFSAR, Section 15.4.
4. 10 CFR 100

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Reactivity Balance

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 element assembly (CEA) 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) - Reactor Trip Breakers Open and LCO 3.1.2, "SHUTDOWN MARGIN (SDM), Reactor Trip Breaker Closed") 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. Reactivity balance is typically based on the 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 variables fixed (such as CEA 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.

(continued)

BASES

BACKGROUND (continued)

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), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

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 critical boron 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

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 CEA withdrawal accidents or CEA 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 calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

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 CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. 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 $\pm$ 1% $\Delta k/k$ 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% $\Delta k/k$ 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.

(continued)

BASES

LCO (continued)

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 MODE 1 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 MODE 2 because enough operating margin exists to limit the affects of a reactivity anomaly and THERMAL POWER is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. This Specification does not apply in MODES 2, 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. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or CEA replacement, or shuffling).

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 restrictions 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% $\Delta k/k$, 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. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

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 CEA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 (continued)

is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. The first Note indicates that the normalization of predicted core reactivity to the measured value may take place 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 (e.g., Azimuthal Tilt) for prompt indication of an anomaly. A Note, "only required after 60 EFPD," is added to the Frequency column to allow this.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. UFSAR, Section 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 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. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups and temperatures 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 (BOC) MTC is less positive than that allowed by the LCO. 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 burnable absorbers, either fixed lumped poison rods or poisons distributed within selected fuel rods to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end of cycle (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 EOC limit.

(continued)

BASES (continued)

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 during accidents, such as overheating and overcooling events.

Reference 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, such as very large soluble boron concentrations, to ensure the accident results are bounding.

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control element assembly (CEA) withdrawal transient from either subcritical or full THERMAL POWER. The limiting overheating event relative to plant response is based on the Loss of Condenser Vacuum event (Ref. 3). The most limiting event with respect to a positive MTC is a CEA withdrawal accident from a subcritical condition, also referred to as a startup accident (Ref. 4).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

LCO 3.1.4 requires the MTC to be within the specified limits of the COLR to ensure 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. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the 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, the limits must also be maintained to ensure accidents, such as the uncontrolled CEA assembly or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES except for a MSLB in MODE 3. In this case, the analysis assumes worst case MTC, with the ECCS systems mitigating the event.

(continued)

BASES (continued)

APPLICABILITY (continued)

However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 days of (before or after) reaching 40 effective full power days and a ⅔ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.

SR 3.1.4.2 is modified by a second Note, which indicates that if extrapolated MTC is more negative than the EOC COLR limit, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. UFSAR, Section 15.0.
3. UFSAR, Section 15.2.
4. UFSAR, Section 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Element Assembly (CEA) Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analyses 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 and GDC 26 (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. If a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

Limits on CEA alignment and operability have been established, and all CEA 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.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately ¾ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS).

(continued)

BASES

BACKGROUND (continued)

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System. Part length CEAs are not credited in the safety analyses for shutting down the reactor, as are the regulating and shutdown groups. The part length CEAs are used solely for ASI control.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Plant Computer CEA Position Indication System and the Reed Switch Position Indication System. The two systems are made up of 3 channels, redundant reed switch assemblies and the pulse control system.

The Plant Computer CEA Position Indication System counts the commands sent to the CEA gripper coils from the CEDMCS that moves the CEAs. There is one step counter for each group of CEAs. Individual CEAs 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 Plant Computer CEA Position Indication System is considered highly precise (± one step or ± ¾ inch). If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on a series of reed switches spaced along a tube with a center to center distance of 1.5 inches, which is two steps, and wired with precision resistors in a voltage divider network. To increase the reliability of the system, there are redundant reed switches at each position. If a CEA slips partially into the core, but does not make up the rod bottom contact logic, the plant computer step counter will incorrectly reflect the CEA position. Therefore, if COLSS is receiving incorrect CEA positions, the operability of the COLSS programs must be determined. Refer to Section 3.2, Power Distribution Limits for the applicable actions.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper.

Inadvertent withdrawal of a single CEA may be caused by an electrical failure in the CEA coil power programmers. A dropped CEA could be caused by an opening of the electrical circuit of the CEDM holding coil for a full length or part length CEA.

The acceptance criteria for addressing CEA inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished. During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one CEA 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 remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The third type of misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in erosion of DNB margin.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Analysis considers the case of a single CEA withdrawn approximately 10 inches from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) bounds the situation when a CEA is misaligned from its group by 6.6 inches.

The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if specified acceptable fuel design limits (SAFDLs) are approached.

Since the CEA drop incidents result in the most rapid approach to SAFDLs caused by a CEA misoperation, the accident analysis analyzed a single full length CEA drop, a single part length CEA drop, and a part length CEA subgroup drop. The most rapid approach to the DNBR SAFDL or the fuel centerline melt SAFDL is caused by a single full length CEA drop.

In the case of the full length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. As the dropped CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped CEA.

For a part length CEA subgroup drop, a distortion in power distribution, and a decrease in core power are produced. As the dropped part length CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to the CPCs, and a reactor trip signal on low DNBR is generated. For the part length CEA drop, both core average power and three dimensional peak to average power density increase promptly. As the dropped part length CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CEA alignment satisfies Criteria 2 and 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The limits on part length, shutdown, and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

The requirement is to maintain the CEA alignment to within 6.6 inches between any CEA and all other CEAs in its group. The minimum misalignment assumed in safety analysis is 9.9 inches, and in some cases, a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, DNBR, and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA 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 (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown modes, the OPERABILITY of the shutdown and regulating CEAs 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 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more CEAs (regulating, shutdown, or part length) are misaligned by 6.6 inches and ≤ 9.9 inches but trippable, or one CEA misaligned by > 9.9 inches but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with the limits in the COLR, and within 2 hours CEA alignment is restored. Regulating and part length CEA alignment can be restored by either aligning the misaligned CEA(s) to within 6.6 inches of its group or aligning the misaligned CEA's group to within 6.6 inches of the misaligned CEA(s). Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within 6.6 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with the limits in the COLR ensures acceptable power distributions are maintained (Ref. 3). For small misalignments (< 9.9 inches) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (≥ 9.9 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small.

Therefore, this condition is limited to the single CEA misalignment, while still allowing 2 hours for recovery.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

In both cases, a 2 hour time period is sufficient to:

a. Identify cause of a misaligned CEA;

b. Take appropriate corrective action to realign the CEAs; and

c. Minimize the effects of xenon redistribution.

The CEA must be returned to OPERABLE status within 2 hours. If a CEA misalignment results in the COLSS programs being declared INOPERABLE, refer to Section 3.2 Power Distribution Limits for applicable actions.

B.1 and B.2

If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8. CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

C.1

If a Required Action or associated Completion Time of Condition A or Condition B is not met, one or more regulating or shutdown CEAs are untrippable, the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

(continued)

BASES

ACTIONS

C.1 (continued)

If a full length CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7, "Regulating Control Element Assembly (CEA)

Insertion Limits," does not ensure that adequate SDM exists. Therefore, the plant must be shut down in order to evaluate the SDM required boron concentration and power level for critical operation. Continued operation is allowed with untrippable part length CEAs if the alignment and insertion limits are met.

Continued operation is not allowed with one or more full length CEAs untrippable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

D.1

Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 9.9 inches. For example, two CEAs in a group misaligned from any other CEA in that group by > 9.9. inches, or more than one CEA group that has a least one CEA misaligned from any other CEA in that group by > 9.9 inches. This is indicative of a loss of power distribution and a loss of safety function, respectively. Multiple CEA misalignments should result in automatic protective action. Therefore, with two or more CEAs misaligned more than 9.9 inches, this could result in a situation outside the design basis and immediate action would be required to prevent any potential fuel damage. Immediately opening the reactor trip breakers minimizes these effects.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual CEA positions are within 6.6 inches (indicated reed switch positions) of all other CEAs in the group at a 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during actual CEA motion, deviations can immediately be detected.

SR 3.1.5.2

OPERABILITY of at least two CEA position indicator channels is required to determine CEA positions, and thereby ensure compliance with the CEA alignment and insertion limits. The CEA full in and full out limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions.

SR 3.1.5.3

Verifying each full length CEA is trippable would require that each CEA be tripped. In MODES 1 and 2 tripping each full length CEA would result in radial or axial power tilts, or oscillations. Therefore individual full length CEAs are exercised every 92 days to provide increased confidence that all full length CEAs continue to be trippable, even if they are not regularly tripped. A movement of 5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one full length CEA is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs (Ref. 3). Between required performances of SR 3.1.5.3, if a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.4

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position. Since this test must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other factors, which determine the OPERABILITY of the CEA Reed Switch Indication System. These factors include:

a. Other, more frequently performed surveillances that help to verify OPERABILITY;

b. On-line diagnostics performed automatically by the CPCs, CEACs, and the Plant Computer which include CEA position comparisons and sensor validation; and

c. The CHANNEL CALIBRATIONs for the CPCs (SR 3.3.1.9) and CEACs (SR 3.3.3.4) input channels that are performed at 18 month intervals and is an overlapping test.

SR 3.1.5.5

Verification of full length CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis (Ref. 7). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures the reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality 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.

The 4 second CEA drop time is the maximum time it takes for a fully withdrawn individual full length CEA to reach its 90% insertion position when electrical power is interrupted to the CEA drive mechanism with RCS T_{cold} greater than or equal to 552°F and all reactor coolant pumps operating.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, 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," and GDC 26, "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 shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAS. The shutdown CEAs must be capable of full withdrawal without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. The shutdown CEAs are controlled manually by the control room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing regulating CEAs during an approach to criticality. The shutdown CEAs are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a CEA ejection accident are limited to acceptable limits.

CEAs are considered fully withdrawn at 144.75 inches, since this position places them outside the active region of the core.

On a reactor trip, all CEAs (shutdown CEAs and regulating CEAs), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." The shutdown CEA insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed") following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, 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 CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA.

The acceptance criteria for addressing shutdown CEA as well as regulating CEA insertion limits and inoperability or misalignment are that:

a. There be no violation of:

1. specified acceptable fuel design limits, or

2. Reactor Coolant System pressure boundary damage integrity; and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The core remains subcritical after accident transients.

The shutdown CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The shutdown CEAs 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.

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODES 1 and 2. The applicability in MODE 2 begins anytime any regulating CEA is not fully inserted. 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. Refer to LCO 3.1.1 and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breaker Closed," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.3, which verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1

Prior to entering this Condition, the shutdown CEAs were fully withdrawn. If a shutdown CEA is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

If the CEA is not within limits, within 2 hours restore the CEA to within limits. The 2 hour total Completion Time allows the operator adequate time to adjust the CEA in an orderly manner and is consistent with the required

(continued)

BASES

ACTIONS (continued)

A.1 (continued)

Completion Times in LCO 3.1.5, "Control Element Assembly (CEA) Alignment."

B.1

When Required Action A.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs (along with the regulating CEAs) 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 CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

Since the shutdown CEAs are positioned manually by the control room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Regulating Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available 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," and GDC 26, "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 regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups generally operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually or automatically controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7; LCO 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within LCO 3.2.1,

(continued)

BASES

BACKGROUND (continued)

"Linear Heat Rate (LHR)"; LCO 3.2.2, "Planar Radial Peaking Factor (F_{xy})"; and LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)," limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident. In addition to the LHR, F_{xy}, and DNBR limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analyses and preserve the minimum required SDM in MODES 1 and 2.

The establishment of limiting safety system settings and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined and a consistent set of radial peaking factors defined. The long term steady state and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses, and provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.7 are specified for the plant, which has been designed for primarily base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

The regulating CEA insertion and alignment limits, ASI and T_q, are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the

(continued)

BASES

BACKGROUND (continued)

event of a CEA ejection accident, and the shutdown and regulating 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 LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, part length CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During CEA misoperation events, 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 a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available in the CEAs to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth CEA remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for MODE 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., Loss of Flow, Seized Rotor, etc. However, the most limiting SDM requirements for MODES 1 and 2 at EOC come from just one transient, Steam Line Break (SLB). The requirements of the SLB event at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger that those at BOC, the available SDMs obtained via the scramming of the CEAs are also substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in the cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has expected shutdown capability. Consequently, adherence to LCOs 3.1.6 and 3.1.7 provides assurance that the available SDMs at any time in cycle will exceed the limiting SDM requirements at that time in the cycle.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Ref. 4).

The regulating CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The limits on regulating CEA 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 CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal, and is imposed to maintain acceptable power peaking during regulating CEA motion. With one or more CEAC's OPERABLE, the COLR provides separate figures for CEA insertion limits with COLSS in service and COLSS out of service.

The power dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

APPLICABILITY

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in

(continued)

BASES

APPLICABILITY (continued)

MODES 3, 4, and 5 is not required, since the power distribution assumptions would not be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO is modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.3. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO. The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 4 and 5) during loss of load events. The requirements of SR 3.1.7.2 for tracking accumulated time between the insertion limits is still applicable following a reactor power cutback operation.

ACTIONS

A.1 and A.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors and may violate input assumptions of the CEA ejection and CEA misoperation events. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1

If the CEAs are inserted between the short term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, peaking factors can develop that are of concern due to Xenon changes (Ref. 4).

(continued)

BASES

ACTIONS

B.1 (continued)

Additionally, since the CEAs can be in this condition without misalignment, penalty factors are not inserted in the core protection calculators to compensate for the developing peaking factors.

Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour, following CEA insertion beyond the short term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure (Ref. 4). The restriction on THERMAL POWER increases shall remain in effect until the Regulating CEA groups are inserted between short term steady state limit and the transient insertion limit for ≤ 4 hours per 24 hour interval. The 15 minute Completion Time ensures that prompt action shall be taken to restrict THERMAL POWER increases.

C.1

With the regulating CEAs inserted between the long term steady state insertion limit and the transient insertion limit, and with the core approaching the 5 effective full power days (EFPD) per 30 EFPD, or 14 EFPD per 365 EFPD limits, the core approaches the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions. In this case, the CEAs must be returned to within the long term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is a reasonable time to return the CEAs to within the long term steady state insertion limits.

The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long term steady state limits, shown on the figures in the COLR, allows sufficient time for borated water to enter the Reactor Coolant System from the chemical addition and makeup systems, and to cause the regulating CEAs to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is

(continued)

BASES

ACTIONS

C.1 (continued)

discovered that the 5 day or 14 day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1

With the PDIL circuit inoperable, performing SR 3.1.7.1 within 1 hour and every 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position every 12 hours, is sufficient to detect CEA positions that may approach the acceptable limits, and provides the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the control room.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.2

Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.

SR 3.1.7.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper CEA alignments.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Regulatory Guide 1.77, Rev. 0, May 1974.
4. UFSAR, Section 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the part length CEAs are initial assumptions in the safety analyses for CEA misoperation events. The insertion limits directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Plants" (Ref. 2). Limits on part length CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is preserved.

The part length CEAs are used for axial power shape control of the reactor. The positions of the part length CEAs are manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8; LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LHR) (LCO 3.2.1, "Linear Heat Rate (LHR)"); planar peaking factor (F_{xy}) (LCO 3.2.2, "Planar Radial Peaking Factors (F_{xy})"); and LCO 3.2.4 limits in the COLR.

Operation within the limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident.

(continued)

BASES

BACKGROUND (continued)

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup; it is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined, and a consistent set of radial peaking factors are defined. The long term (steady state) and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses; they provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.8 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The regulating CEA insertion, part length CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, 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 a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, part length CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The part length CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii). The part length CEAs are required due to the potential peaking factor violations that could occur if part length CEAs exceed insertion limits.

LCO

The limits on part length CEA insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution.

APPLICABILITY

The part length insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution. Applicability in lower MODES is not required, since the power distribution assumptions would not be exceeded in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1, A.2 and B.1

If the part length CEA groups are inserted beyond the transient insertion limit or between the long term (steady state) insertion limit and the transient limit for 7 or more effective full power days (EFPD) out of any 30 EFPD period, or for 14 EFPD or more out of any 365 EFPD period, flux patterns begin to develop that are outside the range assumed for long term fuel burnup. If allowed to continue beyond this limit, the peaking factors assumed as initial conditions in the accident analysis may be invalidated (Ref. 4). Restoring the CEAs to within limits or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures that acceptable peaking factors are maintained.

Since these effects are cumulative, actions are provided to limit the total time the part length CEAs can be out of limits in any 30 EFPD or 365 EFPD period. Since the cumulative out of limit times are in days, an additional Completion Time of 2 hours is reasonable for restoring the part length CEAs to within the allowed limits.

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should commence. A 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 plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of each part length CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and provide the operator with time to undertake the Required Action(s), should insertion limits be found to be exceeded. The 12 hour frequency also takes into account the indication provided by the power dependent insertion limit alarm circuit and other information about CEA group positions available to the operator in the control room.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Regulatory Guide 1.77, Rev. 0, May 1974.
4. UFSAR, Section 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 Special Test Exceptions (STE)-SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine the control element assembly (CEA) worth.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (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 specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

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 design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the

(continued)

BASES

BACKGROUND (continued)

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 all information necessary to permit a detailed execution of 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. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). PHYSICS TESTS for reload fuel cycles are given in Table 1 of ANSI/ANS-19.6.1-1985. Although these PHYSICS TESTS are generally accomplished within the limits of all 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. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed";

b. LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits"; and

c. LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, this LCO places limits on the minimum amount of CEA worth required to be available for reactivity control when CEA worth measurements are performed.

The individual LCOs cited above govern SDM, CEA group height, insertion, and alignment. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature T_c, and pressurizer pressure contribute to maintaining departure from nucleate boiling (DNB) parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accidents are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

SRs are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) be available for trip insertion from the OPERABLE CEAs, provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck CEA. Since LCO 3.1.2 is suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth CEA was stuck out and calculational uncertainties or the estimated highest CEA worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck CEA and subsequent criticality is reduced during this PHYSICS TEST exception by the requirements to determine CEA positions every 2 hours; by the trip of each CEA to be withdrawn within 7 days prior to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

suspending the SDM requirements; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn CEA (Ref. 5).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This STE is required to permit the periodic verification of the actual versus predicted worth of the regulating and shutdown CEAs. The SDM requirements of LCO 3.1.2, the shutdown CEA insertion limits of LCO 3.1.6, and the regulating CEA insertion limits of LCO 3.1.7 may be suspended.

APPLICABILITY

This LCO is applicable in MODES 2 and 3. Although CEA worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3 as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.2.

(continued)

BASES (continued)

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth withdrawn CEA, restoration of the minimum shutdown reactivity requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.2.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification of the position of each partially or fully withdrawn full length or part length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

SR 3.1.9.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The 7 day Frequency ensures that the CEAs are OPERABLE prior to reducing SDM requirements to less than the limits of LCO 3.1.2.

SR 3.1.9.3

During MODE 3, verification that the reactor is subcritical by at least the reactivity equivalent of the highest estimated CEA worth ensures that the minimum negative reactivity requirements are preserved. The negative reactivity requirements are verified by performing a reactivity balance calculation, considering the listed reactivity effects:

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.3 (continued)

a. RCS boron concentration;

b. CEA positions;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

The Frequency of 2 hours is based on the generally slow change in required boron concentration, and it allows sufficient time for the operator to collect the required data.

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.

5. UFSAR, Chapter 14.

6. 10 CFR 50.46.

7. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 Special Test Exceptions (STE)-MODES 1 and 2

BASES

BACKGROUND

The primary purpose of these MODES 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (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 specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

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 design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. 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).

(continued)

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that 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.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all 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. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Control Element Assembly (CEA) Alignment";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits (F_{xy}^{T})";
LCO 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits";
LCO 3.2.2, "Planar Radial Peaking Factors";
LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)";
LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"; and
LCO 3.3.3, "Control Element Assembly Calculators (CEACs)".

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analysis (Ref. 6) places limits on allowable THERMAL POWER during PHYSICS TESTS and requires that the LHR and the departure from nucleate boiling (DNB) parameter be maintained within limits. The power plateau of ≤ 85% RTP and the associated trip setpoints are required to ensure these limits are maintained.

The individual LCOs governing CEA group height, insertion and alignment, ASI, total planar radial peaking factor, total integrated radial peaking factor, and T_q, preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 7). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS, such as those required to:

a. Measure CEA worth;

b. Determine the reactor stability index and damping factor under xenon oscillation conditions;

c. Determine power distributions for nonnormal CEA configurations;

d. Measure rod shadowing factors; and

e. Measure temperature and power coefficients.

Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.1.8, LCO 3.2.2, LCO 3.2.3, LCO 3.2.5 and LCO 3.3.3, may be suspended during the performance of PHYSICS TESTS provided THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP.

APPLICABILITY

This LCO is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to ≤ 85% RTP ensures that LHRs are maintained within acceptable limits.

(continued)

BASES (continued)

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau in MODE 1, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 and B.2

If Required Action A.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and departure from nucleate boiling ratio margins are maintained while LCOs are suspended. The 1 hour Frequency is sufficient, based upon the slow rate of power change and increased operational controls in place during PHYSICS TESTS. Monitoring LHR ensures that the limits are not exceeded.

(continued)

BASES (continued)

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.
5. UFSAR, Chapter 14.
6. UFSAR, Section 15.3.
7. 10 CFR 50.46.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.11 Special Test Exceptions (STE)-Reactivity Coefficient Testing

BASES

BACKGROUND

The primary purpose of Reactivity Coefficient Testing is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine isothermal temperature coefficient, moderator temperature coefficient, and power coefficient.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (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 specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

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 design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation.

(continued)

BASES

BACKGROUND (continued)

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 all information necessary to permit a detailed execution of testing required to ensure that 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.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all 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. As long as the linear heat rate (LHR) and DNBR remain within its limits, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits"; and
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Limits" (LCO 3.4.1.b, RCS Cold Leg Temperature only).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analysis (Ref. 6) requires that the LHR and the departure from nucleate boiling (DNB) parameter be maintained within limits. The associated trip setpoints are required to ensure these limits are maintained.

The individual LCOs governing CEA group height, insertion and alignment, ASI, total planar radial peaking factor, total integrated radial peaking factor, and T_q, preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 7). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO permits Part Length CEAs and Regulating CEAs to be positioned outside of their normal group heights and insertion limits, and RCS cold leg temperature to be outside its limits during the performance of PHYSICS TESTS. These PHYSICS TESTS are required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.7, LCO 3.1.8, and LCO 3.4.1, (for RCS cold leg temperature only) may be suspended during the performance of PHYSICS TESTS provided COLSS is in service.

APPLICABILITY

This LCO is applicable in MODE 1 with THERMAL POWER > 20% RTP because the reactor must be critical at THERMAL POWER levels > 20% RTP to perform the PHYSICS TESTS described in the LCO section.

ACTIONS

A.1

With the LHR or DNBR outside the limits specified in the COLR, adequate safety margin is not assured and power must be reduced to restore LHR and DNBR to within limits. The required Completion Time of 15 minutes ensure prompt action is taken to restore LHR or DNBR to within limits.

(continued)

BASES

ACTIONS (continued)

B.1

When the Required Action cannot be met or completed within the required Completion Time, PHYSICS TEST must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TEST allows the operator sufficient time to change any abnormal conditions back to within the limits of LCO 3.1.7, LCO 3.1.8, and LCO 3.4.1. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

SURVEILLANCE REQUIREMENTS

SR 3.1.11.1

With THERMAL POWER greater than or equal to 20% RTP, LHR and DNBR can be continuously monitored using the COLSS since the COLSS is available with THERMAL POWER above 20% RTP. If COLSS is not available, LHR and DNBR can be continuously monitored using any OPERABLE CPC channel. Continuous monitoring is required to ensure that the LHR and DNBR limits are satisfied at all times. SRs 3.2.1.1 and 3.2.4.1 provide the specific requirements for performing this SR.

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.
5. UFSAR, Chapter 14.
6. UFSAR, Section 15.3.
7. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full or part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the LHR and Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is 1.30 as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers. It is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for specified LHR and DNBR limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

(continued)

BASES

BACKGROUND (continued)

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicate CEA positions. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The LHR and DNBR algorithms are valid within the limits on ASI, F_{xy} and T_q. These limits are obtained directly from initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4).

The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

b. During a loss of 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 DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing the LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

(continued)

BASES

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratios at relatively low core power levels; and

b. As a result of this inaccuracy, the CPCs assume minimum core power of 20% RTP when generating LPD and DNBR trip signals. When core power is below 20% RTP, the core is operating well below its thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1

Operation at or below the COLSS calculated power limit based on the LHR ensures that the LHR limit is not exceeded. If the COLSS calculated core power limit based on the LHR exceeds the operating limit, restoring the LHR to within limit in 1 hour ensures that prompt action is taken to reduce LHR to below the specified limit. One hour is a reasonable time to return LHR to within limits when the limit is exceeded without a trip due to events such as a dropped CEA or an axial xenon oscillation.

B.1, B.2.1, and B.2.2

If the COLSS is not available the OPERABLE LPD channels are monitored to ensure that the LHR limit is not exceeded. Operation within this limit ensures that in the event of a LOCA the fuel cladding temperature does not exceed 2200°F. Four hours is allowed for restoring the LHR limit to within the region of acceptable operation. This duration is reasonable because the COLSS allows the plant to operate with less LHR margin (closer to the LHR limit than when monitoring the CPCs).

(continued)

BASES

ACTIONS

B.1, B.2.1, and B.2.2. (continued)

When operating with the COLSS out of service and LHR not within the region of acceptable operation, there is a possibility of a slow undetectable transient that degrades the LHR slowly over the 4 hour period and is then followed by an AOO or an accident. To remedy this, the CPC calculated values of LHR are monitored every 15 minutes when the COLSS is out of service and LHR not within the region of acceptable operation. The 15 minute frequency is adequate to allow the operator to identify an adverse trend in conditions that could result in an approach to the LHR limit. Also, a maximum allowable change in the CPC calculated LHR ensures that further degradation requires the operators to take immediate action to restore LHR to within limits or reduce reactor power to comply with the Technical Specifications (TS). With an adverse trend, one hour is allowed for restoring LHR to within limits if the COLSS is not restored to OPERABLE status. Implementation of this requirement ensures that reductions in core thermal margin are quickly detected, and if necessary, results in a decrease in reactor power and subsequent compliance with the existing COLSS out of service TS limits. If LHR cannot be monitored every 15 minutes, assume that there is an adverse trend.

With no adverse trend, four hours is allowed to restore the LHR to within limits if the COLSS is not restored to OPERABLE status. This duration is reasonable because the Frequency of the CPC determination of LHR is increased and if operation is maintained steady, the likelihood of exceeding the LHR limit during this period is not increased. The likelihood of induced reactor transients from an early power reduction is also decreased.

C.1

If the LHR cannot be returned to within its limit or the LHR cannot be determined because of the COLSS and CPC inoperability, core power must be reduced. Reduction of core power to $\leq$ 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP.

(continued)

BASES

ACTIONS

C.1 (continued)

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

With the COLSS out of service, the operator must monitor the LHR with any OPERABLE local power density channel. A 2 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LHR limits.

This SR is modified by a Note that states that the SR is applicable only when the COLSS is out of service. Continuous monitoring of the LHR is provided by the COLSS, which calculates core power and core power operating limits based on the LHR and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on LHR. This SR is also modified by a Note that states that the SR is not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. During plant startup (increase from 15-18% RTP), the plant dynamics associated with the downcomer to economizer swapover may result in a temporary power increase above 20% RTP. The 2 hours after reaching 20% RTP is required for plant stabilization.

SR 3.2.1.2

Verification that the COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on the LHR in units of kilowatts per foot ensures the operator is alerted when conditions approach the LHR operating limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 (continued)

The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

REFERENCES

1. UFSAR, Section 15.
2. UFSAR, Section 6.
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.
6. Regulatory Guide 1.77, Rev. 0, May 1974.
7. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Planar Radial Peaking Factors (F_{xy})

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could 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 accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full or part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. Limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution. Power distribution is a product of multiple parameters, various combinations of

(continued)

BACKGROUND (continued)

which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on Linear Heat Rate (LHR) and Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is 1.30 as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR values. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

DNBR penalty factors are included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

The COLSS indicates continuously to the operator how far the core is to the operating limits and provides an audible

(continued)

BASES

BACKGROUND (continued)

alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of 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 DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_q during normal operation. However, fuel cladding damage results if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cause increased power peaking and correspondingly increased LHR.

F_{xy} satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

Limiting the calculated Planar Radial Peaking Factors (F^C_{xy}) used in the COLSS and CPCs to values greater than the measured Planar Radial Peaking Factors (F^M_{xy}) ensures that the limits calculated by the COLSS and CPCs remain valid.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate because of the poor signal to noise ratio that they experience at relatively low core power levels; and

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below 20% RTP, the core is operating well below its thermal limits, and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1.1 and A.1.2

When the F^M_{xy} values exceed the F^C_{xy} values used in the COLSS and CPCs, nonconservative operating limits and trip setpoints may be calculated. In this case, action must be taken to ensure that the COLSS operating limits and CPC trip

(continued)

ACTIONS

A.1.1 and A.1.2 (continued)

setpoints remain valid with respect to the accident analysis. The operator can do this by performing the Required Actions A.1.1 and A.1.2. The 6 hour Completion Time provides the time required to calculate the required multipliers and make the necessary adjustments to the CPC addressable constants. During this period the DNBR and LHR setpoints may be slightly nonconservative but DNBR and LHR are still within limits. Therefore, 6 hours is an acceptable Completion Time to perform these actions considering the low probability of an accident occurring during this time period.

A.2

As an alternative to Required Actions A.1.1 and A.1.2, the operator may adjust the affected values of F^C_{xy} used in the COLSS and CPCs to values $\geq F^H_{xy}$. The 6 hour Completion Time provides the time required to calculate the required multipliers and make the necessary adjustments to the CPC addressable constants. During this period the DNBR and LHR setpoints may be slightly nonconservative but DNBR and LHR are still within limits. Therefore, 6 hours is an acceptable Completion Time to perform these actions considering the low probability of an accident occurring during this time period.

A.3

If Required Actions A.1.1 and A.1.2 or A.2 cannot be accomplished within 6 hours, the core power must be reduced. Reduction to 20% RTP or less ensures that the core is operating within the specified thermal limits and places the core in a conservative condition based on the trip setpoints generated by the COLSS and CPC operating limits; these limits are established assuming a minimum core power of 20% RTP. Six hours is a reasonable time to reach 20% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

This periodic Surveillance is for determining, using the Incore Detector System, that F^M_{xy} values are $\leq F^C_{xy}$ values used in the COLSS and CPCs. It ensures that the F^C_{xy} values used remain valid throughout the fuel cycle. A Frequency of 31 EFPD is acceptable because the power distribution changes only slightly with the amount of fuel burnup. Determining the F^M_{xy} values after each fuel loading when THERMAL POWER is > 40% RTP, but prior to its exceeding 70% RTP, ensures that the core is properly loaded.

REFERENCES

1. UFSAR, Section 15.
2. UFSAR, Section 6.
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.
6. Regulatory Guide 1.77, Rev. 0, May 1974.
7. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AZIMUTHAL POWER TILT (T_q)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could 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 accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full or part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions, (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is 1.30 as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by the assembly. Fuel assemblies that incur higher than average burnup experience greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins caused by the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux data, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detection system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on the ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation and AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of 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 DNB condition (Ref. 4);

c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs due to initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

T_q satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

The limitations on the T_q are provided to ensure that design operating margins are maintained. T_q greater than the limit in the COLR with COLSS in service or > 0.03 with COLSS out of service is not expected. If it occurs, the actions to be taken ensure that operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.

(continued)

BASES

APPLICABILITY (continued)

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating LPD and DNBR trip signals. When the core power is below this level, the core is operating well below its thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1 and A.2

If the measured T_q is greater than the T_q allowance used in the CPCs but within the limit in the COLR with COLSS in service or ≤ 0.03 with COLSS out of service, nonconservative trip setpoints may be calculated. Required Action A.1 restores T_q to within its specified limits by repositioning the CEAs, and the reactor may return to normal operation. A Completion Time of 2 hours is sufficient time to allow the operator to reposition the CEAs because significant radial xenon redistribution does not occur within this time.

If the T_q cannot be restored within 2 hours, the T_q allowance in the CPCs must be adjusted, per Required Action A.2, to be equal to or greater than the measured value of T_q to ensure that the design safety margins are maintained. The COLSS T_q alarm must also be adjusted to the new CPC allowance, so that the COLSS T_q alarm is still valid.

B.1, B.2, B.3, B.4, and B.5

Required Actions B.1, B.2, B.3, B.4, and B.5 are modified by a Note that requires action B.5 be performed if power reduction commences prior to restoring T_q within the limit. This requirement ensures that corrective action is taken before unrestricted power operation resumes.

If the measured T_q is not within the limit in the COLR with COLSS in service or > 0.03 with COLSS out of service, THERMAL POWER is reduced to ≤ 50% RTP within 4 hours. The 4 hours allows enough time to take action to restore T_q prior to reducing power and limits the probability of operation with a power distribution out of limits. Such actions include performing SR 3.2.3.2, which provides a value of T_q that can be used in subsequent actions.

(continued)

ACTIONS

B.1, B.2, B.3, B.4, and B.5 (continued)

Also in the case of a tilt generated by a CEA misalignment, the 4 hours allows recovery of the CEA misalignment, because a measured T_q not within the limit in the COLR with COLSS in service or > 0.03 with COLSS out of service is not expected. If it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. Operation then is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial power peaking factors used in the core power distribution calculation are based on an untilted power distribution.

If the measured T_q is not restored to within its specified limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation, which results in increased LHGRs when the xenon redistributes. If the measured T_q cannot be restored to within its limit within 4 hours, reactor power must be reduced. Reducing THERMAL POWER to < 50% RTP within 4 hours provides an acceptable level of protection from increased power peaking due to potential xenon redistribution while maintaining a power level sufficiently high enough to allow the tilt to be analyzed.

The Variable Overpower trip setpoints are reduced to ≤ 55% RTP to ensure that the assumptions of the accident analysis regarding power peaking are maintained. After power has been reduced to ≤ 50% RTP, the rate and magnitude of changes in the core flux are greatly reduced. Therefore, 16 hours is an acceptable time period to allow for reduction of the Variable Overpower trip setpoints, Required Action B.2. The 16 hour Completion Time allowed to reduce the Variable Overpower trip setpoints is required to perform the actions necessary to reset the trip setpoints.

THERMAL POWER is restricted to 50% RTP until the measured T_q is restored to within its specified limit by correcting the out of limit condition. This action prevents the operator from increasing THERMAL POWER above the conservative limit when a significant T_q has existed, but allows the unit to continue operation for diagnostic purposes.

(continued)

BASES

ACTIONS

B.1, B.2, B.3, B.4, and B.5 (continued)

If T_q is restored prior to identifying and correcting the cause, the plant corrective action program will continue to evaluate the cause of the out of limit condition.

After a THERMAL POWER increase following restoration of T_q, operation may proceed provided the measured T_q is determined to remain within its specified limit at the increased THERMAL POWER level.

The provision to allow discontinuation of the Surveillance after verifying that T_q is within its specified limit at least once per hour for 12 hours or until T_q is verified to be within its specified limit at a THERMAL POWER $\geq$ 95% RTP provides an acceptable exit from this action after the measured T_q has been returned to an acceptable value.

C.1

If the measured T_q cannot be restored or determined within its specified limit, core power must be reduced. Reduction of core power to $\leq$ 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP. Six hours is a reasonable time to reach 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Continuous monitoring of the measured T_q by the incore nuclear detectors is provided by the COLSS. A COLSS alarm is annunciated in the event that the measured T_q exceeds the value used in the CPCs.

With the COLSS out of service, the operator must calculate T_q and verify that it is within its specified limits. The 12 hour Frequency is sufficient to identify slowly developing T_q's before they exceed the limits of this LCO. Also, the 12 hour Frequency prevents significant xenon redistribution.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

This SR is also modified by a Note that states that the SR is not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. During plant startup (increase from 15-18% RTP), the plant dynamics associated with the downcomer to economizer swapover may result in a temporary power increase above 20% RTP. The 2 hours after reaching 20% RTP is required for plant stabilization.

SR 3.2.3.2

Verification that the COLSS T_q alarm actuates at a value less than the value used in the CPCs ensures that the operator is alerted if T_q approaches its operating limit. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

SR 3.2.3.3

Independent confirmation of the validity of the COLSS calculated T_q ensures that the COLSS accurately identifies T_q's. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

(continued)

BASES

REFERENCES

1. UFSAR, Section 15.
2. UFSAR, Section 6.
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.
6. Regulatory Guide 1.77, Rev. 0, May 1974.
7. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial value assumed in the accident analyses. Specifically, operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could 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 accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full or part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analysis (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is 1.30 as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bows and grid spacers and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limits Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and LPD for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience a greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm when an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded during AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of 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 DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

DNBR satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

With the COLSS in service and one or both of the Control Element Assembly Calculators (CEACs) OPERABLE, the DNBR will be maintained by ensuring that the core power calculated by the COLSS is equal to or less than the permissible core power operating limit based on DNBR calculated by the COLSS. In the event that the COLSS is in service but neither of the two CEACs is OPERABLE, the DNBR is maintained by ensuring that the core power calculated by the COLSS is equal to or less than a reduced value of the permissible core power operating limit calculated by the COLSS. In this condition, the calculated operating limit must be reduced by the allowance specified in the COLR.

In instances for which the COLSS is out of service and either one or both of the CEACs are OPERABLE, the DNBR is maintained by operating within the acceptable region specified in the COLR and using any OPERABLE CPC channel. Alternatively, when the COLSS is out of service and neither of the two CEACs is OPERABLE, the DNBR is maintained by operating within the acceptable region specified in the COLR for this condition and using any OPERABLE CPC channel.

(continued)

BASES

LCO (continued)

With the COLSS out of service, the limitation on DNBR as a function of the ASI represents a conservative envelope of operating conditions consistent with the analysis assumptions that have been analytically demonstrated adequate to maintain an acceptable minimum DNBR for all AOOs. Operation of the core with a DNBR at or above this limit ensures that an acceptable minimum DNBR is maintained in the event of the most limiting AOO (i.e., loss of flow transient, CEA misoperation events, or asymmetric SG transient).

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the Local Power Density (LPD) and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1

Operating at or above the minimum required value of the DNBR ensures that an acceptable minimum DNBR is maintained in the event of a postulated AOO. If the core power as calculated by the COLSS exceeds the core power limit calculated by the COLSS based on the DNBR, fuel design limits may not be maintained following an AOO and prompt action must be taken to restore the DNBR above its minimum Allowable Value. With the COLSS in service, 1 hour is a reasonable time for the operator to initiate corrective actions to restore the DNBR above its specified limit, because of the low probability of a severe transient occurring in this relatively short time.

(continued)

BASES

ACTIONS (continued)

B.1, B.2.1, and B.2.2

If the COLSS is not available the OPERABLE DNBR channels are monitored to ensure that the DNBR is not exceeded. Maintaining the DNBR within this specified range ensures that no postulated accident results in consequences more severe than those described in the UFSAR, Chapter 15. A 4 hour Frequency is allowed to restore the DNBR limit to within the region of acceptable operation. This Frequency is reasonable because the COLSS allows the plant to operate with less DNBR margin (closer to the DNBR limit) than when monitoring with the CPCs.

When operating with the COLSS out of service and DNBR outside the region of acceptable operation, there is a possibility of a slow undetectable transient that degrades the DNBR slowly over the 4 hour period and is then followed by an anticipated operational occurrence or an accident. To remedy this, the CPC calculated values of DNBR are monitored every 15 minutes when the COLSS is out of service and DNBR outside the region of acceptable operation. The 15 minute frequency is adequate to allow the operator to identify an adverse trend in conditions that could result in an approach to the DNBR limit. Also, a maximum allowable change in the CPC calculated DNBR ensures that further degradation requires the operators to take immediate action to restore DNBR to within limits or reduce reactor power to comply with the Technical Specifications (TS). With an adverse trend, 1 hour is allowed for restoring DNBR to within limits if the COLSS is not restored to OPERABLE status. Implementation of this requirement ensures that reductions in core thermal margin are quickly detected and, if necessary, results in a decrease in reactor power and subsequent compliance with the existing COLSS out of service TS limits. If DNBR cannot be monitored every 15 minutes, assume that there is an adverse trend.

With no adverse trend, 4 hours is allowed for restoring the DNBR to within limits if the COLSS is not restored to OPERABLE status. This duration is reasonable because the Frequency of the CPC determination of DNBR has been increased, and, if operation is maintained steady, the likelihood of exceeding the DNBR limit during this period is not increased. The likelihood of induced reactor transients from an early power reduction is also decreased.

(continued)

BASES

ACTIONS (continued)

C.1

If the DNBR cannot be restored or determined within the allowed times of Conditions A and B, core power must be reduced. Reduction of core power to ≤ 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 20% RTP from full power conditions in an orderly manner and without challenging plant systems.

SR 3.2.4.1

With the COLSS out of service, the operator must monitor the DNBR as indicated on any of the OPERABLE DNBR channels of the CPCs to verify that the DNBR is within the specified limits shown in the COLR. A 2 hour Frequency is adequate to allow the operator to identify trends in conditions that would result in an approach to the DNBR limit.

This SR is modified by a Note that states that the SR is only applicable when the COLSS is out of service. Continuous monitoring of the DNBR is provided by the COLSS, which calculates core power and core power operating limits based on the DNBR and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on the DNBR. This SR is also modified by a Note that states that the SR is not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. During plant startup (increase from 15-18% RTP), the plant dynamics associated with the downcomer to economizer swapover may result in a temporary power increase above 20% RTP. The 2 hours after reaching 20% RTP is required for plant stabilization.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

Verification that the COLSS margin alarm actuates at a power level equal to or less than the core power operating limit, as calculated by the COLSS, based on the DNBR, ensures that the operator is alerted when operating conditions approach the DNBR operating limit. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Chapter 6.
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.
6. Regulatory Guide 1.77, Rev. 0, May 1974.
7. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could 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 accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the axial power distribution include:

a. Using full or part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the axial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is 1.30 as calculated by the CE-1 Correlation (Ref. 3), and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculators (CPCs). The COLSS and CPCs monitor the core power distribution and are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPC is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of 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 DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis (Ref.1).

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ASI satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The COLR provides separate limits that are based on different combinations of COLSS and CEACs being in and out of service.

The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analyses. The ASI limits ensure that with T_q at its maximum upper limit, the DNBR does not drop below the DNBR Safety Limit for AOOs.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are strongly conservative.

(continued)

BASES

ACTIONS

A.1

The ASI limits specified in the COLR ensure that the LOCA and loss of flow accident criteria assumed in the accident analyses remain valid. If the ASI exceeds its limit, a Completion Time of 2 hours is allowed to restore the ASI to within its specified limit. This duration gives the operator sufficient time to reposition the regulating or part length CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is significantly reduced if the condition is not allowed to persist for more than 2 hours.

B.1

If the ASI is not restored to within its specified limits within the required Completion Time, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased LHGRs when the xenon redistributes. Reducing thermal power to ≤ 20% RTP reduces the maximum LHR to a value that does not exceed the fuel design limits if a design basis event occurs. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. The COLSS provides the operator with an alarm if an ASI limit is approached.

Verification of the ASI every 12 hours ensures that the operator is aware of changes in the ASI as they develop. A 12 hour Frequency for this Surveillance is acceptable because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow changes in the ASI, which can be discovered before the limits are exceeded.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.5.1 (continued)

This SR is modified by a Note that states that the SR is not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. During plant startup (increase from 15-18% RTP), the plant dynamics associated with the downcomer to economizer swapover may result in a temporary power increase above 20% RTP. The 2 hours after reaching 20% RTP is required for plant stabilization.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Chapter 6.
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.
6. Regulatory Guide 1.77, Rev. 0, May 1974.
7. 10 CFR 50, Appendix A, GDC 26.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System (RCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of

(continued)

BASES

BACKGROUND (continued)

occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic; and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic and RTCBs are addressed in LCO 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation." The CEACs are addressed in LCO 3.3.3, "Control Element Assembly Calculators (CEACs)."

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation, the core protection calculators (CPCs), and the CEACs, though complex, are considered components in the measurement channels of the Variable Over Power – High, Logarithmic Power Level – High, DNBR – Low, and Local Power Density (LPD) – High trips.

Four identical measurement channels, designated channels A through D, with electrical and physical separation, are provided for each parameter used in the generation of trip signals, with the exception of the control element assembly (CEA) position indication used in the CPCs. Each measurement channel provides input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features Actuation System (ESFAS)

(continued)

BASES

BACKGROUND

Measurement Channels (continued)

bistables, and most provide indication in the control room. Measurement channels used as an input to the RPS are not used for control functions.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping bistables monitoring the same parameter in two or more channels will de-energize Matrix Logic, which in turn de-energizes the Initiation Logic. This causes all four RTCBs to open, interrupting power to the CEAs, allowing them to fall into the core.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of 10 CFR 50, Appendix A, GDC 21 (Ref. 1). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a channel inoperable, no single additional failure in the RPS can either cause an inadvertent trip or prevent a required trip from occurring.

Adequate channel to channel independence includes physical and electrical independence of each channel from the others. This allows operation in two-out-of-three logic with one channel removed from service until following the next MODE 5 entry. Since no single failure will either cause or prevent a protective system actuation, and no protective channel feeds a control, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. 4).

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. Four separate CPCs perform the calculations independently, one for each of the four RPS channels. The CPCs provide outputs to drive display indications (DNBR margin, LPD margin, and calibrated neutron flux power levels) and provide DNBR – Low and LPD – High pretrip and trip signals. The CPC channel outputs for the DNBR – Low and LPD – High trips operate contacts in the Matrix Logic in a manner identical to the other RPS trips.

(continued)

BASES

BACKGROUND

Measurement Channels (continued)

Each CPC receives the following inputs:

- Hot leg and cold leg temperatures;
- Pressurizer pressure;
- Reactor coolant pump speed;
- Excore neutron flux levels;
- Target CEA positions; and
- CEAC penalty factors.

Each CPC is programmed with "addressable constants." These are various alignment values, correction factors, etc., that are required for the CPC computations. They can be accessed for display or for the purpose of changing them as necessary.

The CPCs use this constant and variable information to perform a number of calculations. These include the calculation of CEA group and subgroup deviations (and the assignment of conservative penalty factors), correction and calculation of average axial power distribution (APD) (based on excore flux levels and CEA positions), calculation of coolant flow (based on pump speed), and calculation of calibrated average power level (based on excore flux levels and ΔT power).

The DNBR calculation considers primary pressure, inlet temperature, coolant flow, average power, APD, radial peaking factors, and CEA deviation penalty factors from the CEACs to calculate the state of the limiting (hot) coolant channel in the core. A DNBR–Low trip occurs when the calculated value reaches the minimum DNBR trip setpoint.

(continued)

BASES

BACKGROUND

Measurement Channels (continued)

The LPD calculation considers APD, average power, radial peaking factors (based upon target CEA position), and CEAC penalty factors to calculate the current value of compensated peak power density. An LPD–High trip occurs when the calculated value reaches the trip setpoint. The four CPC channels provide input to the four DNBR–Low and four LPD–High RPS trip channels. They effectively act as the sensor (using many inputs) for these trips.

The CEACs perform the calculations required to determine the position of CEAs within their subgroups for the CPCs. Two independent CEACs compare the position of each CEA to its subgroup position. If a deviation is detected by either CEAC, an annunciator sounds and appropriate "penalty factors" are transmitted to all CPCs. These penalty factors conservatively adjust the effective operating margins to the DNBR–Low and LPD–High trips. Each CEAC also drives a single cathode ray tube (CRT), which is switchable between CEACs. The CRT displays individual CEA positions from the selected CEAC.

Each CEA has two separate reed switch assemblies mounted outside the RCPB. Each of the two CEACs receives CEA position input from one of the two reed switch position transmitters on each CEA, so that the position of all CEAs is independently monitored by both CEACs.

CEACs are addressed in LCO 3.3.3.

Bistable Trip Units

Bistable trip units, mounted in the Plant Protection System (PPS) cabinet, receive an analog input from the measurement channels. They compare the analog input to trip setpoints and provide contact output to the Matrix Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A, B, C, and D, for each RPS parameter, one for each measurement channel. Bistables de-energize when a trip occurs, in turn de-energizing bistable relays mounted in the PPS relay card racks.

(continued)

BASES

BACKGROUND

Bistable Trip Units (continued)

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

Some measurement channels provide contact outputs to the PPS. In these cases, there is no bistable card, and opening the contact input directly de-energizes the associated bistable relays. These include the CPC generated DNBR – Low and LPD – High trips.

The trip setpoints used in the bistables are based on the analytical limits derived from the accident analysis (Ref. 5). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), Allowable Values specified in Table 3.3.1-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 7). The nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances. A channel is inoperable if its actual setpoint is not within its Allowable Value.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational, and processor uncertainties and dynamic allowances as defined in the latest applicable revision of CEN-PSD-335-P, "Functional Design Requirements for a Core Protection Calculation" (Ref. 10) and CEN-PSD-336-P," Functional Design Requirements for a Control Element Assembly Calculator," (Ref. 11).

(continued)

BASES

BACKGROUND

Bistable Trip Units (continued)

Setpoints in accordance with the Allowable Value will ensure that SLs of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during AOOs, and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Note that in LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

Functional testing of the entire RPS, from bistable input through the opening of individual RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. Nuclear instrumentation, the CPCs, and the CEACs can be similarly tested. UFSAR, Section 7.2 (Ref. 8), provides more detail on RPS testing. Processing transmitter calibration is normally performed on a refueling basis.

RPS Logic

The RPS Logic, addressed in LCO 3.3.4, consists of both Matrix and Initiation Logic and employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic.

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB initiation relays. The trip paths thus each have six contacts in series, one from each matrix, and perform a

(continued)

BASES

BACKGROUND

RPS Logic (continued)

logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one of the four RTCBs. The RTCB initiation relays, when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four initiation relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all four RTCBs, tripping them open.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable relay cards, up to but not including the matrix relays. Matrix contacts on the bistable relay cards are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, initiation relays, and the initiation relay contacts in the RTCB control circuitry.

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BASES

BACKGROUND

RPS Logic (continued)

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition.

Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

Two-out-of-three logic also prevents inadvertent trips caused by any single channel failure in a trip condition.

In addition to the trip channel bypasses, there are also operating bypasses on select RPS trips. These bypasses are enabled manually in all four RPS channels when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Operating bypasses are normally implemented in the bistable, so that normal trip indication is also disabled. Trips with operating bypasses include Pressurizer Pressure – Low, Logarithmic Power Level – High, and CPC (DNBR – Low and LPD – High).

Reactor Trip Circuit Breakers (RTCBs)

The reactor trip switchgear, addressed in LCO 3.3.4, consists of four RTCBs. Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel, such that the loss of either MG set does not de-energize the CEDMs. Power is supplied from the MG sets to the CEDM's via two redundant paths (trip legs). Trip legs 1 and 3 are in parallel with Trip legs 2 and 4. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

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BASES

BACKGROUND

Reactor Trip Circuit Breakers (RTCBs) (continued)

Each of the two trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

Each RTCB is operated by either a manual reactor trip push button, a Supplementary Protection System (SPS) trip relay or an RPS actuated Initiation relay. There are four Manual Trip push buttons each push button operates one of the four RTCBs. Depressing either of the push buttons in both trip legs will result in a reactor trip.

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and Initiation relays are not utilized, and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to the RTCBs necessary to actuate both the undervoltage and shunt trip attachments but excludes the Initiation relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. UFSAR, Section 7.2 (Ref. 8), explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Design Basis Definition

The RPS is designed to ensure that the following operational criteria are met:

- The associated actuation will occur when the parameter monitored by each channel reaches its setpoint and the specific coincidence logic is satisfied;
- Separation and redundancy are maintained to permit a channel to be out of service for testing or maintenance while still maintaining redundancy within the RPS instrumentation network.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis takes credit for most of the RPS trip Functions. Those functions for which no credit is taken, termed equipment protective functions, are not needed from a safety perspective.

Each RPS setpoint is chosen to be consistent with the function of the respective trip. The basis for each trip setpoint falls into one of three general categories:

Category 1: To ensure that the SLs are not exceeded during AOOs;

Category 2: To assist the ESFAS during accidents; and

Category 3: To prevent material damage to major plant components (equipment protective).

The RPS maintains the SLs during AOOs and mitigates the consequences of DBAs in all MODES in which the RTCBs are closed.

Each of the analyzed transients and accidents can be detected by one or more RPS Functions. Functions not specifically credited in the accident analysis are part of the NRC staff approved licensing basis for the plant. Noncredited Functions include the Steam Generator #1 Level – High, and the Steam Generator #2 Level – High. These trips minimize the potential for equipment damage.

The specific safety analysis applicable to each protective function are identified below:

1. Variable Over Power-High

The Variable Over Power - High Trip (VOPT) is provided to protect the reactor core during positive reactivity addition excursions. Under steady state conditions the trip setpoint will stay above the neutron power level signal by a preset value, called the band function. When the power level increases the setpoint will increase to attempt to maintain the separation defined by the Band function, however the rate of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

1. Variable Over Power-High (continued)

setpoint change is limited by the rate function. If the power level signal increases faster than the setpoint, a trip will occur when the power level eventually equals the trip setpoint. The maximum value the setpoint can have is determined by the ceiling function.

The Variable Over Power – High trip provides protection against core damage during the following events:

- Uncontrolled CEA Withdrawal From Low Power (AOO);
- Uncontrolled CEA Withdrawal at Power (AOO); and
- CEA Ejection (Accident).

2. Logarithmic Power Level – High

The Logarithmic Power Level – High trip protects the integrity of the fuel cladding and helps protect the RCPB in the event of an unplanned criticality from a shutdown condition.

In MODES 2, 3, 4, and 5, with the RTCBs closed and the CEA Drive System capable of CEA withdrawal, protection is required for CEA withdrawal events originating when THERMAL POWER is < 1E-4% RTP. For events originating above this power level, other trips provide adequate protection.

MODES 3, 4, and 5, with the RTCBs closed, are addressed in LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown."

In MODES 3, 4, or 5, with the RTCBs open or the CEAs not capable of withdrawal, the Logarithmic Power Level – High trip does not have to be OPERABLE. The indication and alarm functions required to indicate a boron dilution event are addressed in LCO 3.3.12, "Boron Dilution Alarm System (BDAS)".

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

3. Pressurizer Pressure – High

The Pressurizer Pressure – High trip provides protection for the high RCS pressure SL. In conjunction with the pressurizer safety valves and the main steam safety valves (MSSVs), it provides protection against overpressurization of the RCPB during the following events:

- Loss of Condenser Vacuum (AOO);
- CEA Withdrawal From Low Power Conditions (AOO);
- Chemical and Volume Control System Malfunction (AOO); and
- Main Feedwater System Pipe Break (Accident).

4. Pressurizer Pressure - Low

The Pressurizer Pressure – Low trip is provided to trip the reactor to assist the ESF System in the event of loss of coolant accidents (LOCAs). During a LOCA, the SIs may be exceeded; however, the consequences of the accident will be acceptable. A Safety Injection Actuation Signal (SIAS) and a Containment Isolation Actuation Signal (CIAS) are initiated simultaneously.

5. Containment Pressure – High

The Containment Pressure – High trip prevents exceeding the containment design pressure psig during a design basis LOCA or main steam line break (MSLB) accident. During a LOCA or MSLB the SLs may be exceeded; however, the consequences of the accident will be acceptable. An SIAS, CIAS, and MSIS are initiated simultaneously.

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BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

6, 7. Steam Generator Pressure – Low

The Steam Generator #1 Pressure – Low and Steam Generator #2 Pressure – Low trips provide protection against an excessive rate of heat extraction from the steam generators and resulting rapid, uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESF System in the event of an MSLB or main feedwater line break accident. A main steam isolation signal (MSIS) is initiated simultaneously.

8, 9. Steam Generator Level – Low

The Steam Generator #1 Level – Low and Steam Generator #2 Level – Low trips ensure that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the RCS due to the loss of the heat sink:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (AOO);
- Loss of Condenser Vacuum (AOO);
- Loss of Normal Feedwater Event (AOO);
- Feedwater System Pipe Break (Accident); and
- Single RCP Rotor Seizure (AOO)

10, 11. Steam Generator Level – High

The Steam Generator #1 Level – High and Steam Generator #2 Level – High trips are provided to protect the turbine from excessive moisture carryover in case of a steam generator overfill event. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

12, 13. Reactor Coolant Flow – Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event and a large steam line break with concurrent loss of offsite AC power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to prevent violation of the peak linear heat rate or DNBR Safety Limits.

14. Local Power Density – High

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The LPD – High trip provides protection against fuel centerline melting due to the occurrence of excessive local power density peaks during the following AOOs:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to the steam line rupture) Without Turbine Trip;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; and
- CEA Misoperation; Single Part Length CEA Drop.

For the events listed above (except CEA Misoperation; Single Part Length CEA Drop), DNBR – Low will trip the reactor first, since DNB would occur before fuel centerline melting would occur.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR – Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power;
- CEA Misoperation; Full Length CEA Drop;
- CEA Misoperation; Part Length CEA Subgroup Drop;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

In the above list, only the steam generator tube rupture, the RCP shaft seizure, and the sample or instrument line break are accidents. The rest are AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

15. Departure from Nucleate Boiling Ratio (DNBR)-Low (continued)

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

PARAMETER	LIMITING VALUE
RCS Cold Leg Temperature - Low	≥ 505°F
RCS Cold Leg Temperature - High	≤ 590°F
Axial Shape Index - Positive	Not more positive than +0.5
Axial Shape Index - Negative	Not more negative than -0.5
Pressurizer Pressure - Low	≥ 1860 psia
Pressurizer Pressure - High	≤ 2388 psia
Integrated Radial Peaking Factor - Low	≥ 1.28
Integrated Radial Peaking Factor - High	≤ 7.00
Quality Margin - Low	> 0

Interlocks/Bypasses

The operating bypasses and their Allowable Values are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific Table entries.

The automatic operating bypass removal features must function as a backup to manual actions for all safety related trips to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The basis for each of the operating bypasses is discussed under individual trips in the LCO section:

a. Logarithmic Power Level – High;

b. DNBR – Low and LPD – High.

The RPS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Actions allow maintenance (trip channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed. With one channel in each Function trip channel bypassed, this effectively places the plant in a two-out-of-three logic configuration in those Functions.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 7).

The Bases for the individual Function requirements are as follows:

1. Variable Over Power-High

   This LCO requires all four channels of Variable Over Power High to be OPERABLE in MODES 1 and 2.

   The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Variable Over Power High reactor trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA ejection accident occur.

(continued)

BASES

LCO

2. Logarithmic Power Level – High

This LCO requires all four channels of Logarithmic Power Level – High to be OPERABLE in MODE 2.

In MODES 3, 4, or 5 when the RTCBs are shut and the CEA Drive System is capable of CEA withdrawal conditions are addressed in LCO 3.3.2.

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Logarithmic Power Level – High reactor trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA withdrawal event occur.

The Logarithmic Power Level – High trip may be bypassed when THERMAL POWER is above 1E-4% RTP to allow the reactor to be brought to power during a reactor startup. This operating bypass is automatically removed when THERMAL POWER decreases below 1E-4% RTP. Above 1E-4% RTP, the Variable Over Power – High and Pressurizer Pressure – High trips provide protection for reactivity transients.

3. Pressurizer Pressure – High

This LCO requires four channels of Pressurizer Pressure – High to be OPERABLE in MODES 1 and 2.

The Allowable Value is set below the nominal lift setting of the pressurizer code safety valves, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a loss of condenser vacuum at 100% power, this setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS and consequent pressure rise. The pressurizer safety valves may lift to prevent overpressurization of the RCS.

(continued)

BASES

LCO

4. Pressurizer Pressure – Low (continued)

This LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODES 1 and 2.

The Allowable Value is set low enough to prevent a reactor trip during normal plant operation and pressurizer pressure transients. However, the setpoint is high enough that with a LOCA, the reactor trip will occur soon enough to allow the ESF systems to perform as expected in the analyses and mitigate the consequences of the accident.

5. Containment Pressure – High

The LCO requires four channels of Containment Pressure – High to be OPERABLE in MODES 1 and 2.

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. It is set low enough to initiate a reactor trip when an abnormal condition is indicated.

6, 7. Steam Generator Pressure – Low

This LCO requires four channels of Steam Generator #1 Pressure – Low and Steam Generator #2 Pressure – Low to be OPERABLE in MODES 1 and 2.

This Allowable Value is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down, resulting in positive reactivity addition to the core. If the moderator temperature coefficient is negative a reactor trip is required to offset that effect.

(continued)

BASES

LCO

6. 7. Steam Generator Pressure – Low (continued)

The trip setpoint may be manually decreased as steam generator pressure is reduced during controlled plant cooldown, provided the margin between steam generator pressure and the setpoint is maintained ≤ 200 psia. This allows for controlled depressurization of the secondary system while still maintaining an active reactor trip setpoint and MSIS setpoint, until the time is reached when the setpoints are no longer needed to protect the plant. The setpoint increases automatically as steam generator pressure increases until the specified trip setpoint is reached.

8. 9. Steam Generator Level – Low

This LCO requires four channels of Steam Generator #1 Level – Low and Steam Generator #2 Level – Low for each steam generator to be OPERABLE in MODES 1 and 2. The Allowable Value is sufficiently below the normal operating level for the steam generators so as not to cause a reactor trip during normal plant operations. The input signal providing the reactor trip input also provides an input to a bistable that initiates auxiliary feedwater to the affected generator via the Auxiliary Feedwater Actuation Signal (AFAS). The trip setpoint ensures that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling. The reactor trip will remove the heat source (except decay heat), thereby conserving the reactor heat sink.

10. 11. Steam Generator Level – High

This LCO requires four channels of Steam Generator #1 Level – High and Steam Generator #2 Level – High to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to allow for normal plant operation and transients without causing a reactor trip. It is set low enough to ensure a reactor trip occurs before the level reaches the steam dryers. Having steam generator water level at the trip value is indicative of the plant not being operated in a controlled manner.

(continued)

BASES

12, 13. Reactor Coolant Flow – Low

This LCO requires four channels of Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator # 2-Low to be OPERABLE in MODES 1 and 2. The Allowable Value is set low enough to allow for slight variations in reactor coolant flow during normal plant operations while providing the required protection. Tripping the reactor ensures that the resultant power to flow ratio provides adequate core cooling to maintain DNBR under the expected pressure conditions for this event.

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," ensure adequate RCS flow rate is maintained.

14. Local Power Density – High

This LCO requires four channels of LPD – High to be OPERABLE.

The LCO on the CPCs ensures that the SLs are maintained during all AOOs and the consequences of accidents are acceptable.

A CPC is not considered inoperable if CEAC inputs to the CPC are inoperable. The Required Actions required in the event of CEAC channel failures ensure the CPCs are capable of performing their safety Function.

The CPC channels may be manually bypassed below 1E-4% RTP, as sensed by the logarithmic nuclear instrumentation. This bypass is enabled manually in all four CPC channels when plant conditions do not warrant the trip protection. The bypass effectively removes the DNBR – Low and LPD – High trips from the RPS Logic circuitry. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied.

(continued)

BASES

LCO

14. Local Power Density – High (continued)

This operating bypass is required to perform a plant startup, since both CPC generated trips will be in effect whenever shutdown CEAs are inserted. It also allows system tests at low power with Pressurizer Pressure – Low or RCPs off.

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

This LCO requires four channels of DNBR – Low to be OPERABLE.

The LCO on the CPCs ensures that the SLs are maintained during all AOOs and the consequences of accidents are acceptable.

A CPC is not considered inoperable if CEAC inputs to the CPC are inoperable. The Required Actions required in the event of CEAC channel failures ensure the CPCs are capable of performing their safety Function.

The CPC channels may be manually bypassed below 1E-4% RTP, as sensed by the logarithmic nuclear instrumentation. This bypass is enabled manually in all four CPC channels when plant conditions do not warrant the trip protection. The bypass effectively removes the DNBR – Low and LPD – High trips from the RPS logic circuitry. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied.

This operating bypass is required to perform a plant startup, since both CPC generated trips will be in effect whenever shutdown CEAs are inserted. It also allows system tests at low power with Pressurizer Pressure – Low or RCPs off.

(continued)

BASES

LCO

Interlocks/Bypasses (continued)

The LCO on operating bypass permissive removal channels requires that the automatic operating bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODES addressed in the specific LCO for each Function. All four bypass removal channels must be OPERABLE to ensure that none of the four RPS channels are inadvertently bypassed.

This LCO applies to the operating bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue. In the case of the Logarithmic Power Level – High trip (Function 2), the absence of a bypass will limit maximum power to below the trip setpoint.

The interlock function Allowable Values are based upon analysis of functional requirements for the bypassed Functions. These are discussed above as part of the LCO discussion for the affected Functions.

APPLICABILITY

This LCO is applicable to the RPS Instrumentation in MODES 1 and 2. LCO 3.3.2 is applicable to the RPS Instrumentation in MODES 3, 4, and 5 with any RTCB closed and any CEA capable of withdrawal. The requirements for the CEACs in MODES 1 and 2 are addressed in LCO 3.3.3. The RPS Matrix Logic, Initiation Logic, RTCBs, and Manual Trips in MODES 1, 2, 3, 4, and 5 are addressed in LCO 3.3.4.

Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The reactor trips are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the ESFAS in providing acceptable consequences during accidents. Most trips are not required to be OPERABLE in MODES 3, 4, and 5. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM. Exceptions to this are:

- The Logarithmic Power Level – High trip, RPS Logic RTCBs, and Manual Trip are required in MODES 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events.

(continued)

BASES

APPLICABILITY (continued)

- Steam Generator Pressure-Low trip, is required in MODE 3, with the RTCBs closed to provide protection for steam line break events in MODE 3.

The Logarithmic Power Level – High trip, and the Steam Generator Pressure-Low trip in these lower MODES are addressed in LCO 3.3.2. The Logarithmic Power Level – High trip is bypassed prior to MODE 1 entry and is not required in MODE 1.

ACTIONS

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, then all affected functions provided by that channel must be declared inoperable, and the unit must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS (continued)

One Note has been added to the ACTIONS. Note 1 has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit (Bypassed or Tripped)
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density-High (RPS) DNBR-Low (RPS)
2. Pressurizer Pressure-High (Narrow Range)	Pressurizer Pressure-High (RPS) Local Power Density-High (RPS) DNBR-Low (RPS)
3. Steam Generator Pressure-Low	Steam Generator Pressure-Low (RPS) Steam Generator #1 Level-Low (ESF) Steam Generator #2 Level-Low (ESF)
4. Steam Generator Level-Low (Wide Range)	Steam Generator Level-Low (RPS) Steam Generator #1 Level-Low (ESF) Steam Generator #2 Level-Low (ESF)
5. Core Protection Calculator	Local Power Density-High (RPS) DNBR-Low (RPS)

A.1 and A.2

Condition A applies to the failure of a single trip channel or associated instrument channel inoperable in any RPS automatic trip Function. RPS coincidence logic is two-out-of-four.

(continued)

BASES

ACTIONS (continued)

If one RPS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip in 1 hour (Required Action A.1). The 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable. The failed channel must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. With a channel in bypass, the coincidence logic is now in a two-out-of-three configuration.

The Completion Time of prior to entering MODE 2 following the next MODE 5 entry is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip.

B.1

Condition B applies to the failure of two channels in any RPS automatic trip Function.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES, even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS is in a two-out-of-three logic; but with another channel failed, the RPS may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip.

(continued)

BASES

ACTIONS

B.1 (continued)

This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One of the two inoperable channels will need to be restored to operable status prior to the next required CHANNEL FUNCTIONAL TEST, because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not possible to bypass more than one RPS channel, and placing a second channel in trip will result in a reactor trip. Therefore, if one RPS channel is in trip and a second channel is in bypass, a third inoperable channel would place the unit in LCO 3.0.3.

C.1, C.2.1, and C.2.2

Condition C applies to one automatic bypass removal channel inoperable. If the inoperable operating bypass removal channel for any operating bypass channel cannot be restored to OPERABLE status within 1 hour, the associated RPS channel may be considered OPERABLE only if the operating bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the affected automatic trip channel placed in maintenance (trip channel) bypass or trip. The operating bypass removal channel and the automatic trip channel must be repaired prior to entering MODE 2 following the next MODE 5 entry. The Bases for the Required Actions and required Completion Times are consistent with Condition A.

D.1 and D.2

Condition D applies to two inoperable automatic operating bypass removal channels. If the operating bypass removal channels for two operating bypasses cannot be restored to OPERABLE status within 1 hour, the associated RPS channel may be considered OPERABLE only if the operating bypass is not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the operating bypass either removed or one automatic trip channel placed in maintenance (trip channel) bypass and the other in trip within 1 hour.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

The restoration of one affected bypassed automatic trip channel must be completed prior to the next CHANNEL FUNCTIONAL TEST, or the plant must shut down per LCO 3.0.3 as explained in Condition B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

E.1

Condition E applies if any CPC cabinet receives a high temperature alarm. There are redundant temperature sensors in each of the four CPC bays. Since CPC bays B and C also house CEAC calculators 1 and 2, respectively, a high temperature in either of these bays requires entry into LCO 3.3.3, Condition C.

If a CPC cabinet high temperature alarm is received, it is possible for an OPERABLE CPC to be affected and not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on OPERABLE CPCs within 12 hours. The Completion Time of 12 hours is adequate considering the low probability of undetected failure, the consequences of a single channel failure, and the time required to perform a CHANNEL FUNCTIONAL TEST.

F.1

Condition F applies if an OPERABLE CPC has three or more autorestarts in a 12 hour period.

CPCs and CEACs will attempt to autorestart if they detect a fault condition, such as a calculator malfunction or loss of power. A successful autorestart restores the calculator to operation; however, excessive autorestarts might be indicative of a calculator problem.

(continued)

BASES

ACTIONS

F.1 (continued)

If a nonbypassed CPC has three or more autorestarts, it may not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on the CPC to ensure it is functioning properly. Based on plant operating experience, the Completion Time of 24 hours is adequate and reasonable to perform the test while still keeping the risk of operating in this condition at an acceptable level, since overt channel failure will most likely be indicated and annunciated in the control room by CPC online diagnostics.

G.1

Condition G is entered when the Required Action and associated Completion Time of Condition A, B, C, D, E, or F are not met.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Time, the reactor must be brought to a MODE where the Required Actions do not apply. 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.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

In the case of RPS trips with multiple inputs, such as the DNBR and LPD inputs to the CPCs, a CHANNEL CHECK must be performed on all inputs.

SR 3.3.1.2

The RCS flow rate indicated by each CPC is verified, as required by a Note, to be less than or equal to the actual RCS total flow rate, determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations, every 12 hours when THERMAL POWER is ≥ 70% RTP. The 12 hours after reaching 70% RTP is for plant stabilization, data taking, and flow verification. This check (and if necessary, the adjustment of the CPC addressable constant flow coefficients) ensures that the DNBR setpoint is conservatively adjusted with respect to actual flow indications, as determined by the Core Operating Limits Supervisory System (COLSS).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

SR 3.3.1.3

The CPC autorestart count is checked every 12 hours to monitor the CPC and CEAC for normal operation. If three or more autorestarts of a nonbypassed CPC occur within a 12 hour period, the CPC may not be completely reliable. Therefore, the Required Action of Condition F must be performed. The auto restart periodic tests restart (Code 30) and normal system load (Code 33) are not included in this total. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 12 hour interval.

SR 3.3.1.4

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 20%. The Linear Power Level signal and the CPC addressable constant multipliers are adjusted to make the CPC ΔT power and nuclear power calculations agree with the calorimetric calculation if the absolute difference is ≥ 2% when THERMAL POWER is ≥ 80% RTP, and -0.5% to 10% when THERMAL POWER is between 20% and 80%. The value of 2% when THERMAL POWER is ≥ 80% RTP, and -0.5% to 10% when THERMAL POWER is between 20% and 80% is adequate because this value is assumed in the safety analysis. These checks (and, if necessary, the adjustment of the Linear Power Level signal and the CPC addressable constant coefficients) are adequate to ensure that the accuracy of these CPC calculations is maintained within the analyzed error margins. The power level must be > 20% RTP to obtain accurate data. At lower power levels, the accuracy of calorimetric data is questionable.

The tolerance between 20% and 80% RTP is +10% to reduce the number of adjustments required as the power level increases. The -0.5% tolerance between 20% and 80% RTP is based on the reduced accuracy of the calorimetric data inputs at low power levels. Performing a calorimetric

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4 (continued)

calibration with a -0.5% tolerance at low power levels ensures the difference will remain within -2.0% when power is increased above 80% RTP. If a calorimetric calculation is performed above 80% RTP, it will use accurate inputs to the calorimetric calculation available at higher power levels. When the power level is decreased below 80% RTP an additional performance of the SR to the -0.5% to 10% tolerance is not required if the SR has been performed above 80% RTP.

The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs. The Frequency is modified by a Note indicating this Surveillance need only be performed within 12 hours after reaching 20% RTP.

The 12 hours after reaching 20% RTP is required for plant stabilization, data taking, and flow verification. The secondary calorimetric is inaccurate at lower power levels. A second Note in the SR indicates the SR may be suspended during PHYSICS TESTS. The conditional suspension of the daily calibrations under strict administrative control is necessary to allow special testing to occur.

SR 3.3.1.5

The RCS flow rate indicated by each CPC is verified to be less than or equal to the RCS total flow rate every 31 days. The Note indicates the Surveillance is performed within 12 hours after THERMAL POWER is ≥ 70% RTP. This check (and, if necessary, the adjustment of the CPC addressable flow constant coefficients) ensures that the DNBR setpoint is conservatively adjusted with respect to actual flow indications as determined either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or by a calorimetric calculation. Operating experience has shown the specified Frequency is adequate, as instrument drift is minimal and changes in actual flow rate are minimal over core life.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

The three vertically mounted excore nuclear instrumentation detectors in each channel are used to determine APD for use in the DNBR and LPD calculations. Because the detectors are mounted outside the reactor vessel, a portion of the signal from each detector is from core sections not adjacent to the detector. This is termed shape annealing and is compensated for after every refueling by performing SR 3.3.1.11, which adjusts the gains of the three detector amplifiers for shape annealing. SR 3.3.1.6 ensures that the preassigned gains are still proper. When power is < 15% the CPCs do not use the excore generated signals for axial flux shape information. The Note allowing 12 hours after reaching 15% RTP is required for plant stabilization and testing. The 31 day Frequency is adequate because the demonstrated long term drift of the instrument channels is minimal.

SR 3.3.1.7

A CHANNEL FUNCTIONAL TEST on each channel is performed every 92 days to ensure the entire channel will perform its intended function when needed. The SR is modified by two Notes. Note 1 is a requirement to verify the correct CPC addressable constant values are installed in the CPCs when the CPC CHANNEL FUNCTIONAL TEST is performed. Note 2 allows the CHANNEL FUNCTIONAL TEST for the Logarithmic Power Level – High channels to be performed 2 hours after power drops below 1E-4% RTP.

The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 8. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Bistable Tests

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. This is done with the affected RPS channel trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the interval between surveillance interval extension analysis.

The requirements for this review are outlined in Reference 9.

Matrix Logic Tests

Matrix Logic tests are addressed in LCO 3.3.4. This test is performed one matrix at a time. It verifies that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts.

Trip Path Tests

Trip path (Initiation Logic) tests are addressed in LCO 3.3.4. These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, thereby opening the affected RTCB. The RTCB must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 9).

The CPC and CEAC channels and excore nuclear instrumentation channels are tested separately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests (continued)

The excore channels use preassigned test signals to verify proper channel alignment. The excore logarithmic channel test signal is inserted into the preamplifier input, so as to test the first active element downstream of the detector.

The power range excore test signal is inserted at the drawer input, since there is no preamplifier.

The quarterly CPC CHANNEL FUNCTIONAL TEST is performed using software. This software includes preassigned addressable constant values that may differ from the current values.

Provisions are made to store the addressable constant values on a computer disk prior to testing and to reload them after testing. A Note is added to the Surveillance Requirements to verify that the CPC CHANNEL FUNCTIONAL TEST includes the correct values of addressable constants.

SR 3.3.1.8

A Note indicates that neutron detectors are excluded from CHANNEL CALIBRATION. A CHANNEL CALIBRATION of the power range neutron flux channels every 92 days ensures that the channels are reading accurately and within tolerance (Ref. 9). The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the interval between surveillance interval extension analysis. The requirements for this review are outlined in Reference 9. Operating experience has shown this Frequency to be satisfactory. The detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8 (continued)

meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4) and the monthly linear subchannel gain check (SR 3.3.1.6). In addition, the associated control room indications are monitored by the operators.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a CHANNEL CALIBRATION every 18 months.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 9.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis as well as operating experience and consistency with the typical 18 month fuel cycle.

The Surveillance is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4) and the monthly linear subchannel gain check (SR 3.3.1.6).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

Every 18 months, a CHANNEL FUNCTIONAL TEST is performed on the CPCs. The CHANNEL FUNCTIONAL TEST shall include the injection of a signal as close to the sensors as practicable to verify OPERABILITY including alarm and trip Functions.

The basis for the 18 month Frequency is that the CPCs perform a continuous self monitoring function that eliminates the need for frequent CHANNEL FUNCTIONAL TESTS. This CHANNEL FUNCTIONAL TEST essentially validates the self monitoring function and checks for a small set of failure modes that are undetectable by the self monitoring function. Operating experience has shown that undetected CPC or CEAC failures do not occur in any given 18 month interval.

SR 3.3.1.11

The three excore detectors used by each CPC channel for axial flux distribution information are far enough from the core to be exposed to flux from all heights in the core, although it is desired that they only read their particular level. The CPCs adjust for this flux overlap by using the predetermined shape annealing matrix elements in the CPC software.

After refueling, it is necessary to re-establish or verify the shape annealing matrix elements for the excore detectors based on more accurate incore detector readings. This is necessary because refueling could possibly produce a significant change in the shape annealing matrix coefficients.

Incore detectors are inaccurate at low power levels. THERMAL POWER should be significant but < 70% to perform an accurate axial shape calculation used to derive the shape annealing matrix elements.

By restricting power to ≤ 70% until shape annealing matrix elements are verified, excessive local power peaks within the fuel are avoided. Operating experience has shown this Frequency to be acceptable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.12

SR 3.3.1.12 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.7, except SR 3.3.1.12 is applicable only to operating bypass functions and is performed once within 92 days prior to each startup. Proper operation of operating bypass permissives is critical during plant startup because the operating bypasses must be in place to allow startup operation and must be automatically removed at the appropriate points during power ascent to enable certain reactor trips. Consequently, the appropriate time to verify operating bypass removal function OPERABILITY is just prior to startup. The allowance to conduct this Surveillance within 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 9). Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.7. Therefore, further testing of the operating bypass function after startup is unnecessary.

SR 3.3.1.13

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. Response times are conducted on an 18 month STAGGERED TEST BASIS. This results in the interval between successive surveillances of a given channel of n x 18 months, where n is the number of channels in the function. The Frequency of 18 months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power, since equipment operation is required. Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.13 (continued)

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 21.
2. 10 CFR 100.
3. NRC Safety Evaluation Report, July 15, 1994.
4. IEEE Standard 279-1971, April 5, 1972.
5. UFSAR, Chapters 6 and 15.
6. 10 CFR 50.49.
7. "Calculation of Trip Setpoint Values, Plant Protection System". CEN-286(v), or Calculation 13-JC-SG-203 for the Low Steam Generator Pressure Trip function.
8. UFSAR, Section 7.2.
9. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989, and Calculation 13-JC-SB-200.
10. CEN-PSD-335-P, "Functional Design Requirements for a Core Protection Calculator."
11. CEN-PSD-336-P, "Functional Design Requirements for a Control Element Assembly Calculator."

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protective System (RPS) Instrumentation – Shutdown

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and reactor coolant pressure boundary (RCPB) integrity during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an (continued) accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND
(continued)

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic; and
- Reactor trip circuit breakers (RTCBs).

This LCO applies to the Logarithmic Power Level – High trip in MODES 3, 4, and 5 with the RTCBs closed and the CEAs capable of withdrawal. In MODES 1 and 2, this trip function is addressed in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation – Operating." LCO 3.3.12, "Boron Dilution Alarm System (BDAS)," applies when the RTCBs are open.

This LCO applies to the Steam Generator #1 and the Steam Generator #2 Pressure-Low trip in MODE 3, with the RTCBs closed and the CEAs capable of withdrawal. In MODES 1 and 2, this trip function is addressed in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation-Operating."

(continued)

BASES

BACKGROUND (continued)

Measurement Channels and Bistable Trip Units

The measurement channels providing input to the Logarithmic Power Level – High trip consist of the four logarithmic nuclear instrumentation channels detecting neutron flux leakage from the reactor vessel. Other aspects of the Logarithmic Power Level – High trip are similar to the other measurement channels and bistables. These are addressed in the Background section of LCO 3.3.1.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. Nuclear instrumentation can be similarly tested. UFSAR, Section 7.2 (Ref. 3), provides more detail on RPS testing.

APPLICABLE SAFETY ANALYSES

The RPS functions to maintain the SLs during AOOs and mitigates the consequence of DBAs in all MODES in which the RTCBs are closed.

Each of the analyzed transients and accidents can be detected by one or more RPS Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. Noncredited Functions include the Steam Generator Water Level - High Trip. The Steam Generator Water Level - High Trip is purely equipment protective, and its use minimizes the potential for equipment damage.

The Logarithmic Power Level – High trip protects the integrity of the fuel cladding and helps protect the RCPB in the event of an unplanned criticality from a shutdown condition.

The Steam Generator Pressure-Low trip function provides shutdown margin to prevent or minimize the return to power, following a large Main Steam Line Break (MSLB) in MODE 3.

With less than 4 RCPs running the trip setpoint for the Logarithmic Power Level-High trip is reduced to $\leq 10^{-4}$% RTP. The lower setpoint is required for a bank CEA withdrawal with less than 4 RCPs running.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In MODES 2, 3, 4, and 5, with the RTCBs closed, and the Control Element Assembly (CEA) Drive System capable of CEA withdrawal, protection is required for CEA withdrawal events, and excessive cooldown due to a MSLB originating when THERMAL POWER is < 1E-4% RTP. For events originating above this power level, other trips provide adequate protection.

MODES 3, 4, and 5, with the RTCBs closed, are addressed in this LCO. MODE 2 is addressed in LCO 3.3.1.

In MODES 3, 4, or 5, with the RTCBs open or the CEAs not capable of withdrawal, the Logarithmic Power Level – High trip does not have to be OPERABLE. The indication and alarm functions required to indicate a boron dilution event are addressed in LCO 3.3.12 "Boron Dilution Alarm System (BDAS)".

Interlock/Bypasses

The operating bypasses and their Allowable Values are addressed in footnotes to Table 3.3.2-1. They are not otherwise addressed as specific Table entries.

The automatic operating bypass removal features must function as a backup to manual actions for all safety related trips to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The basis for the Logarithmic Power Level -High operating bypass is discussed under individual trips in the LCO section.

The RPS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The LCO requires the Logarithmic Power Level – High, the Steam Generator #1 Pressure-Low, and the Steam Generator #2 Pressure-Low, RPS Functions to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Function.

Actions allow maintenance (trip channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed. With one channel in each Function trip channel bypassed, this effectively places the plant in a two-out-of-three logic configuration in those Functions.

Only the Allowable Values are specified for this RPS trip Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoint is selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function.

These uncertainties are defined in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 4). A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

This LCO requires all four channels of the Logarithmic Power Level – High to be OPERABLE MODES in 3, 4, or 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal.

A CEA is considered capable of withdrawal when power is applied to the Control Element Drive Mechanisms (CEDMs). There are several methods used to remove power from the CEDMs, such as de-energizing the CEDM MGs, opening the CEDM MG output breakers, opening the Control Element Assembly Control System (CEDMCS) CEA breakers, opening the RTCBs, or disconnecting the power cables from the CEDMs. Any method

(continued)

BASES

LCO (continued)

that removes power from the CEDMs may be used. The CEAs are still capable of withdrawal if the CEDMCS withdrawal circuits are disabled with power applied to the CEDMs because failures in the CEDMCS could result in CEA withdrawal.

This LCO requires all four channels of Steam Generator #1 Pressure-Low, and Steam Generator #2 Pressure-Low, to be OPERABLE in MODE 3, when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal. These RPS functions are not required in MODES 4 and 5 because the Steam Generator temperature is low, therefore the energy release and resulting cooldown following a large MSLB in MODES 4 and 5 is not significant.

The Allowable Values are high enough to provide an operating envelope that prevents unnecessary Logarithmic Power Level – High reactor trips during normal plant operations. The Allowable Values are low enough for the system to maintain a safety margin for unacceptable fuel cladding damage should a CEA withdrawal or MSLB event occur.

The Logarithmic Power Level – High trip may be bypassed when THERMAL POWER is above 1E-4% RTP to allow the reactor to be brought to power during a reactor startup. This bypass is automatically removed when THERMAL POWER decreases below 1E-4% RTP. Above 1E-4% RTP, the Variable Over Power – High and Pressurizer Pressure – High trips provide protection for reactivity transients.

APPLICABILITY

This LCO is applicable to the RPS Instrumentation in MODES 3, 4, and 5 with any RTCB closed and any CEA capable of withdrawal. LCO 3.3.1 is applicable to the RPS Instrumentation in MODES 1 and 2. The requirements for the CEACs in MODES 1 and 2 are addressed in LCO 3.3.3. The RPS Matrix Logic, Initiation Logic, RTCBs, and Manual Trips in MODES 1, 2, 3, 4, and 5 are addressed in LCO 3.3.4.

Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The trips are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the Engineered Safety Features Actuation System (ESFAS) in providing acceptable consequences during accidents. Most trips are not required to be OPERABLE in MODES 3, 4, and 5.

(continued)

BASES

APPLICABILITY (continued)

In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM. Exceptions to this are:

- The Logarithmic Power Level – High trip, RPS Logic RTCBs, and Manual Trip are required in MODES 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events. The Logarithmic Power Level – High trip in these lower MODES is addressed in this LCO. The RPS Logic in MODES 1, 2, 3, 4, and 5 is addressed in LCO 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation."

- The Steam Generator #1 Pressure-Low, and the Steam Generator #2 Pressure-Low trips, RPS Logic, RTCBs, and Manual Trip are required in MODE 3 with the RTCBs closed, to provide protection for large MSLB events in MODE 3. The Steam Generator Pressure-Low trip in this lower MODE is addressed in this LCO. The RPS Logic in MODES 1,2,3,4, and 5 is addressed in LCO 3.3.4, Reactor Protection System (RPS) Logic and Trip Initiation.

The applicability for the Logarithmic Power Level-High function is modified by a Note that allows the trip to be bypassed when THERMAL POWER is > 1E-4% RTP, and the bypass is automatically removed when THERMAL POWER is ≤ 1E-4% RTP.

ACTIONS

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is less conservative than the Allowable Value stated in the LCO, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

(continued)

BASES

ACTIONS (continued)

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the excore logarithmic power channel or RPS bistable trip unit is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the unit must enter the Condition for the particular protection Function affected.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

PROCESS MEASUREMENT CIRCUIT	FUNCTIONAL UNIT (Bypassed or Tripped)
Steam Generator Pressure-Low	Steam Generator Pressure - Low (RPS) Steam Generator #1 Level - Low (ESF) Steam Generator #2 Level - Low (ESF)

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered, if applicable in the current MODE of operation.

A.1, and A.2

Condition A applies to the failure of a single trip channel or associated instrument channel inoperable in any RPS function.

The RPS coincidence logic is two-out-of-four. If one channel is inoperable, operation in MODES 3, 4, and 5 is allowed to continue, providing the inoperable channel is placed in bypass or trip in 1 hour (Required Action A.1).

The 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel while ensuring that the risk involved in operating with the failed channel is acceptable.

The failed channel must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. With a channel bypassed, the coincidence logic is now in a two-out-of-three configuration. The Completion Time is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip.

(continued)

BASES

ACTIONS (continued)

B.1

Condition B applies to the failure of two trip channels or associated instrument channels, in any RPS automatic trip function. Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels and still ensures the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS is in a two-out-of-three logic; but with another channel failed, the RPS may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One of the two inoperable channels will need to be restored to OPERABLE status prior to the next required CHANNEL FUNCTIONAL TEST because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not possible to bypass more than one RPS channel, and placing a second channel in trip will result in a reactor trip. Therefore, if one RPS channel is in trip and a second channel is in bypass, a third inoperable channel would place the unit in LCO 3.0.3.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

C.1, C.2.1, and C.2.2

Condition C applies to one automatic operating bypass removal channel inoperable. If the operating bypass removal channel for the high logarithmic power level operating bypass cannot be restored to OPERABLE status within 1 hour,

(continued)

BASES

ACTIONS

C.1, C.2.1 and C.2.2 (continued)

the associated RPS channel may be considered OPERABLE only if the operating bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the operating bypass either removed or the affected automatic channel placed in trip or maintenance (trip channel) bypass. Both the operating bypass removal channel and the associated automatic trip channel must be repaired prior to entering MODE 2 following the next MODE 5 entry. The Bases for the Required Actions and required Completion Times are consistent with Condition A.

D.1 and D.2

Condition D applies to two inoperable automatic operating bypass removal channels. If the operating bypass removal channels for two operating bypasses cannot be restored to OPERABLE status within 1 hour, the associated RPS channel may be considered OPERABLE only if the operating bypass is not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the operating bypass either removed or one automatic trip channel placed in maintenance (trip channel) bypass and the other in trip within 1 hour. The restoration of one affected bypassed automatic trip channel must be completed prior to the next CHANNEL FUNCTIONAL TEST or the plant must shut down per LCO 3.0.3, as explained in Condition B. Completion Times are consistent with Condition B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

E.1

Condition E is entered when the Required Actions and associated Completion Times of Condition A, B, C, or D are not met.

(continued)

BASES

ACTIONS

E.1 (continued)

If Required Actions associated with these Conditions cannot be completed within the required Completion Time, all RTCBs must be opened, placing the plant in a condition where the RPS trip channels are not required to be OPERABLE. A Completion Time of 1 hour is a reasonable time to perform the Required Action, which maintains the risk at an acceptable level while having one or two channels inoperable.

SURVEILLANCE REQUIREMENTS

The SR's for any particular RPS function are found in the SR column of Table 3.3.2-1 for that function. The SRs are an extension of those listed in LCO 3.3.1, listed here because of their Applicability in these MODES.

SR 3.3.2.1

SR 3.3.2.1 is the performance of a CHANNEL CHECK of each RPS channel. This SR is identical to SR 3.3.1.1. Only the Applicability differs.

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limits.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1 (continued)

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on each channel, except power range neutron flux, is performed every 92 days to ensure the entire channel will perform its intended function when needed. This SR is identical to SR 3.3.1.7. Only the Applicability differs.

The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in the UFSAR, Section 7.2 (Ref. 3). These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. This is done with the affected RPS channel trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 6.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Matrix Logic Tests

Matrix Logic Tests are addressed in LCO 3.3.4. This test is performed one matrix at a time. It verifies that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts.

Trip Path Test

Trip path (Initiation Logic) tests are addressed in LCO 3.3.4. These tests are similar to the Matrix Logic tests except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 6). The excore channels use preassigned test signals to verify proper channel alignment. The excore logarithmic channel test signal is inserted into the preamplifier input, so as to test the first active element downstream of the detector.

SR 3.3.2.3

SR 3.3.2.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.2.2, except SR 3.3.2.3 is applicable only to operating bypass functions and is performed once within 92 days prior to each startup. This SR is identical to SR 3.3.1.12. Only the Applicability differs.

Proper operation of operating bypass permissives is critical during plant startup because the operating bypasses must be in place to allow startup operation and must be automatically removed at the appropriate points during power ascent to enable certain reactor trips. Consequently, the appropriate time to verify operating bypass removal function

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.3 (continued)

OPERABILITY is just prior to startup. The allowance to conduct this Surveillance within 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 6). Once the operating bypasses are removed, the operating bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.2.2. Therefore, further testing of the operating bypass function after startup is unnecessary.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is identical to SR 3.3.1.9. Only the Applicability differs.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor (the sensor is excluded for the Logarithmic Power Level Function). The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 6.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis and includes operating experience and consistency with the typical 18 month fuel cycle.

The Surveillance is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4).

SR 3.3.2.5

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. Response times are conducted on an 18 month STAGGERED TEST BASIS. This results in the interval between successive tests of a given channel of n x 18 months, where n is the number of channels in the Function. The 18 month Frequency is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power, since equipment operation is required. Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4).

REFERENCES

1. 10 CFR 50.
2. 10 CFR 100.
3. UFSAR, Section 7.2.
4. "Calculation of Trip Setpoint Values Plant Protection System, CEN-286(v)", or Calculation 13-JC-SG-203 for the Low Steam Generator Pressure Trip Function.

(continued)

BASES

REFERENCES (continued)

5. NRC Safety Evaluation Report, July 15, 1994.

6. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989, and Calculation 13-JC-SB-200.

B 3.3 INSTRUMENTATION

B 3.3.3 Control Element Assembly Calculators (CEACs)

BASES

BACKGROUND

The Reactor Protective System (RPS) initiates a reactor trip to protect against violating the core Specified Acceptable Fuel Design Limits (SAFDLs) and breaching the Reactor Coolant Pressure Boundary (RCPB) during Anticipated Operational Occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features Systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS (defined in this Specification as the Allowable Value), in conjunction with the LCOs, establish the thresholds for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a

(continued)

BACKGROUND (continued)

different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic; and
- Reactor Trip Circuit Breakers (RTCBs).

This LCO addresses the CEACs. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation–Operating," provides a description of this equipment in the RPS.

The excore nuclear instrumentation, the Core Protection Calculators (CPCs), and the CEACs are considered components in the measurement channels of the Variable Over Power-High, Logarithmic Power Level–High, DNBR–Low, and Local Power Density (LPD)–High trips. The CEACs are addressed by this Specification.

All four CPCs receive Control Element Assembly (CEA) deviation penalty factors from each CEAC and use the larger of the penalty factors from the two CEACs in the calculation of DNBR and LPD. CPCs are further described in the Background section of LCO 3.3.1.

The CEACs perform the calculations required to determine the position of CEAs within their subgroups for the CPCs. Two independent CEACs compare the position of each CEA to its subgroup position. If a deviation is detected by either CEAC, an annunciator sounds and appropriate "penalty factors" are transmitted to all CPCs. These penalty factors conservatively adjust the effective operating margins to the DNBR–Low and LPD–High trips. Each CEAC also drives a single Cathode Ray Tube (CRT), which is switchable between CEACs. The CRT displays individual CEA positions from the selected CEAC.

(continued)

BASES

BACKGROUND (continued)

Each CEA has two separate reed switch assemblies mounted outside the RCPB. Each of the two CEACs receives CEA position input from one of the two reed switch position transmitters on each CEA, so that the position of all CEAs is independently monitored by both CEACs.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. Nuclear instrumentation, the CPCs, and the CEACs can be similarly tested. UFSAR, Section 7.2 (Ref. 3), provides more detail on RPS testing. Process transmitter calibration is normally performed on a refueling basis.

APPLICABLE SAFETY ANALYSES

Each of the analyzed transients and accidents can be detected by one or more RPS Functions.

The effect of any misoperated CEA within a subgroup on the core power distribution is assessed by the CEACs, and an appropriately augmented power distribution penalty factor will be supplied as input to the CPCs. As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and doppler feedback effects, the CPCs will initiate a DNBR–Low or LPD–High trip signal if SAFDLs are approached. Each CPC also directly monitors one "target CEA" from each subgroup and uses this information to account for excessive radial peaking factors for events involving CEA groups out of sequence and subgroup deviations within a group, without the need for CEACs.

Therefore, although the CEACs do not provide a direct reactor trip Function, their input to the CPCs is taken credit for in the CEA misoperation analysis.

The CEACs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO on the CEACs ensures that the CPCs are either informed of individual CEA position within each subgroup, using one or both CEACs, or that appropriate conservatism is included in the CPC calculations to account for anticipated

(continued)

BASES

LCO (continued)

CEA deviations. Each CEAC provides an identical input into all four CPC channels. Each CPC uses the higher of the two CEAC transmitted CEA deviation penalty factors. Thus, only one OPERABLE CEAC is required to provide CEA deviation protection. This LCO requires both CEACs to be OPERABLE so that no single CEAC failure can prevent a required reactor trip from occurring.

APPLICABILITY

This LCO is applicable to the CEACs in MODES 1 and 2. The RPS Instrumentation in MODES 1 and 2 is addressed in LCO 3.3.1. The RPS Instrumentation in MODES 3, 4, and 5 with any RTCB closed and any CEA capable of withdrawal is addressed in LCO 3.3.2. The RPS Matrix Logic, Initiation Logic, RTCB, and Manual Trips in Modes 1, 2, 3, 4, and 5 are addressed in LCO 3.3.4.

Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The trips are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the Engineered Safety Features Actuation System in providing acceptable consequences during accidents. Most trips are not required to be OPERABLE in MODES 3, 4, and 5. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

Because CEACs provide the inputs to the DNBR–Low and LPD–High trips, they are required to be OPERABLE in MODES 1 and 2 for the same reasons.

(continued)

BASES

ACTIONS

A.1 and A.2

Condition A applies to the failure of a single CEAC channel. There are only two CEACs, each providing CEA deviation input into all four CPC channels. The CEACs include complex diagnostic software, making it unlikely that a CEAC will fail without informing the CPCs of its failed status. With one failed CEAC, the CPC will receive CEA deviation penalty factors from the remaining OPERABLE CEAC. If the second CEAC should fail (Condition B), the CPC will use large preassigned penalty factors. The specific Required Actions allowed are as follows:

With one CEAC inoperable, the second CEAC still provides a comprehensive set of comparison checks on individual CEAs within subgroups, as well as outputs to all CPCs, CEA deviation alarms, and position indication for display. Verification every 4 hours that each CEA is within 6.6 inches of the other CEAs in its group provides a check on the position of all CEAs and provides verification of the proper operation of the remaining CEAC. An OPERABLE CEAC will not generate penalty factors until deviations of > 6.6 inches within a subgroup are encountered.

The Completion Time of once per 4 hours is adequate based on operating experience, considering the low probability of an undetected CEA deviation coincident with an undetected failure in the remaining CEAC within this limited time frame.

As long as Required Action A.1 is accomplished as specified, the inoperable CEAC can be restored to OPERABLE status within 7 days. The Completion Time of 7 days is adequate for most repairs, while minimizing risk, considering that dropped CEAs are detectable by the redundant CEAC, and other LCOs specify Required Actions necessary to maintain DNBR and LPD margin.

(continued)

BASES

ACTIONS
(continued)

B.1, B.2, B.3, B.4, B.5 and B.6

Condition B applies if the Required Action and associated Completion Time of Required Action A are not met, or if both CEACs are inoperable. Actions associated with this Condition involve disabling the Control Element Drive Mechanism Control System (CEDMCS), while providing increased assurance that CEA deviations are not occurring and informing all OPERABLE CPC channels, via a software flag, that both CEACs are failed. This will ensure that the large penalty factor associated with two CEAC failures will be applied to CPC calculations. The penalty factor for two failed CEACs is sufficiently large that power must be maintained significantly < 100% RTP if CPC generated reactor trips are to be avoided. The Completion Time of 4 hours is adequate to accomplish these actions while minimizing risks.

The Required Actions are as follows:

B.1

Meeting the DNBR margin requirements of LCO 3.2.4, "DNBR" ensures that power level is within a conservative region of operation based on actual core conditions.

B.2

This Action requires that the CEAs are maintained fully withdrawn (≥ 144.75"), except as required for specified testing or flux control via group #5. This verification ensures that undesired perturbations in local fuel burnup are prevented. The Upper Electrical Limit (UEL) CEA reed switches provide an acceptable indication of CEA position.

B.3

The "RSPT/CEAC Inoperable" addressable constant in each of the OPERABLE CPCs is set to indicate that both CEACs are inoperable. This provides a conservative penalty factor to ensure that a conservative effective margin is maintained by the CPCs in the computation of DNBR and LPD trips.

(continued)

BASES

ACTIONS (continued)

B.4

The CEDMCS is placed and maintained in "STANDBY MODE," except during CEA motion permitted by Required Action B.2, to prevent inadvertent motion and possible misalignment of the CEAs.

B.5

A comprehensive set of comparison checks on individual CEAs within groups must be made within 4 hours. Verification that each CEA is within 6.6 inches of other CEAs in its group provides a check that no CEA has deviated from its proper position within the group.

B.6

The Reactor Power Cutback (RPCB) System must be disabled. This ensures that CEA position will not be affected by RPCB operation.

C.1

Condition C applies if the CPC channel B or C cabinet receives a high temperature alarm. There are redundant temperature sensors in each of the four CPC bays. A high temperature alarm in any CPC cabinet requires entry into LCO 3.3.1, Condition E. Since CPC bays B and C also house CEAC calculators 1 and 2, respectively, a high temperature in either of these bays may also indicate a problem with the associated CEAC.

If a CPC channel B or C cabinet high temperature alarm is received, it is possible for an OPERABLE CEAC to be affected and not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on OPERABLE CEACs within 12 hours. The Completion Time of 12 hours is adequate, considering the low probability of undetected failure, the consequences of failure, and the time required to perform a CHANNEL FUNCTIONAL TEST.

(continued)

ACTIONS (continued)

D.1

Condition D applies if an OPERABLE CEAC has three or more auto restarts in a 12 hour period.

CPCs and CEACs will attempt to auto restart if they detect a fault condition such as a calculator malfunction or loss of power. A successful auto restart restores the calculator to operation; however, excessive auto restarts might be indicative of a calculator problem. The auto restart periodic test restart (Code 30), and normal system load (Code 33) are not included in the total.

If an operable CEAC has three or more auto restarts, it may not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on the CEAC to ensure it is functioning properly. Based on plant operating experience, the Completion Time of 24 hours is adequate and reasonable to perform the test while still keeping the risk of operating in this condition at an acceptable level, since overt channel failure will most likely be indicated and annunciated by CPC online diagnostics.

E.1

Condition E is entered when the Required Action and associated Completion Time of Condition B, C, or D are not met.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Time, the reactor must be brought to a MODE where the Required Actions do not apply. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limits.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels.

The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

The CEAC auto restart count is checked every 12 hours to monitor the CPC and CEAC for normal operation. If three or more auto restarts of a nonbypassed CPC occur within a 12 hour period, the CPC may not be completely reliable. The autorestart periodic test restart (code 30) and normal system load (code 33) are not included in the total. Therefore, the Required Action of Condition D must be performed. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 12 hour interval.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.3

A CHANNEL FUNCTIONAL TEST on each CEAC channel is performed every 92 days to ensure the entire channel will perform its intended function when needed. The quarterly CHANNEL FUNCTIONAL TEST is performed using test software. The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

SR 3.3.3.4

SR 3.3.3.4 is the performance of a CHANNEL CALIBRATION every 18 months.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillance. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 5.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis and includes operating experience and consistency with the typical 18 month fuel cycle.

SR 3.3.3.5

Every 18 months, a CHANNEL FUNCTIONAL TEST is performed on the CEACs. The CHANNEL FUNCTIONAL TEST shall include the injection of a signal as close to the sensors as practicable to verify OPERABILITY, including alarm and trip Functions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.5 (continued)

The basis for the 18 month Frequency is that the CEACs perform a continuous self monitoring function that eliminates the need for frequent CHANNEL FUNCTIONAL TESTS. This CHANNEL FUNCTIONAL TEST essentially validates the self monitoring function and checks for a small set of failure modes that are undetectable by the self monitoring function. Operating experience has shown that undetected CPC or CEAC failures do not occur in any given 18 month interval.

SR 3.3.3.6

The isolation characteristics of each CEAC CEA position isolation amplifier are verified once per refueling to ensure that a fault in a CEAC or a CPC channel will not render another CEAC or CPC channel inoperable. The CEAC CEA position isolation amplifiers, mounted in CPC cabinets A and D, prevent a CEAC fault from propagating back to CPC A or D.

The CEA position Isolation amplifier isolation characteristics test shall include the following; 1) with 120 VAC (60 HZ) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 VDC, with an applied input voltage of 5-10 VDC, and 2) with 120 VAC (60 HZ) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 VDC.

The Frequency is based on plant operating experience with regard to channel OPERABILITY, which demonstrates the failure of a channel in any 18 month interval is rare.

REFERENCES

1. 10 CFR 50.
2. 10 CFR 100.
3. UFSAR, Section 7.2.
4. NRC Safety Evaluation Report, July 15, 1994
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989, and Calculation 13-JC-SB-200.

B 3.3 INSTRUMENTATION

B 3.3.4 Reactor Protective System (RPS) Logic and Trip Initiation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of

(continued)

BASES

BACKGROUND (continued)

occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic; and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses the RPS Logic and RTCBs, including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation–Operating," provides a description of the role of this equipment in the RPS. This is summarized below:

RPS Logic

The RPS Logic, consisting of Matrix and Initiation Logic, employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic.

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path

(continued)

BASES

BACKGROUND

RPS Logic (continued)

provides power to one of the four normally energized RTCB Initiation relays. The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one of the four RTCBs. The RTCB Initiation relays, when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four initiation relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all four RTCBs, tripping them open.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts and interconnecting matrix wiring between bistable relay cards, up to but not including the matrix relays. Matrix contacts on the bistable relay cards are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and initiation relays and the initiation relay contacts in the RTCB control circuitry.

(continued)

BASES

BACKGROUND

RPS Logic (continued)

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

Reactor Trip Circuit Breakers (RTCBs)

The reactor trip switchgear consists of four RTCBs. Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. Power is supplied from the MG sets to the CEDMS via two redundant paths (trip legs). Trip legs 1 and 3 are in parallel with Trip legs 2 and 4. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the two trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

(continued)

BASES

BACKGROUND

Reactor Trip Circuit Breakers (RTCBs) (continued)

Each RTCB is operated by either a Manual Trip push button, a Supplementary Protection System (SPS) Trip relay, or an RPS actuated Initiation relay. There are four Manual Trip push buttons, each of the pushbuttons operates one of the RTCBs. Depressing either of the push buttons in both trip legs will result in a reactor trip.

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and Initiation relays are not utilized and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to the RTCBs necessary to actuate both the undervoltage and shunt trip attachments, but excludes the Initiation relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of the individual RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. UFSAR, Section 7.2 (Ref. 3), explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to maintain the SLs during AOOs and assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

Reactor Trip Circuit Breakers (RTCBs)

All of the transient and accident analyses that call for a reactor trip assume that the RTCBs operate and interrupt power to the CEDMs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Reactor Protective System (RPS) Logic

The LCO on the RPS Logic channels ensures that each of the following requirements are met:

- A reactor trip will be initiated when necessary;
- The required protection system coincidence logic is maintained (minimum two-out-of-three, normal two-out-of-four); and
- Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

Failures of individual bistable relays and their contacts, are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies, or the failure of the trip channel bypass contact in the bypass condition.

A matrix logic is considered inoperable if a coincident trip in the same function in the two OPERABLE channels monitored by the Logic Matrix will not remove power from the coils of all four matrix relays. The OPERABILITY of the Matrix Logic is not affected by bypassed or inoperable measurement channels.

(continued)

BASES

LCO

Reactor Protective System (RPS) Logic (continued)

Loss of a single vital bus will de-energize one of the two power supplies in each of three matrices. This will result in two RTCBs opening; however, the remaining two closed RTCBs will prevent a reactor trip. For the purposes of this LCO, de-energizing up to three matrix power supplies due to a single failure is to be treated as a single channel failure, providing the affected matrix relays de-energize as designed, opening the affected RTCBs.

Each of the four Initiation Logic channels opens one RTCB if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Each Initiation Logic channel has its own power supply and is independent of the others. An Initiation Logic channel includes the matrix relay through to the Initiation relay contacts, which open the RTCB.

An Initiation Logic is considered inoperable if the contacts on the initiation relay will not operate when power is removed from the coils of any of the six matrix relays in the trip paths.

It is possible for two Initiation Logic channels affecting the same trip leg to de-energize if a matrix power supply or vital instrument bus fails. This will result in opening the two affected RTCBs.

If one RTCB has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected RTCB may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant RTCB will provide protection if a trip should be required. It is unlikely that a trip will be required during the Surveillance, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or vital bus failure has opened two RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any RTCBs are closed and any CEA is capable of being withdrawn.

(continued)

BASES

LCO (continued)

2. Initiation Logic

This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when the RTCBs are closed and any CEA is capable of being withdrawn.

3. Reactor Trip Circuit Breakers

The LCO requires four RTCB channels to be OPERABLE in MODES 1 and 2, as well as in MODES 3, 4, and 5 when the RTCBs are closed and any CEA is capable of being withdrawn.

Each channel consists of a breaker operated by the Initiation Logic or Manual Trip circuitry.

Without reliable RTCBs and associated support circuitry, a reactor trip cannot occur whether initiated automatically or manually.

Each channel of RTCBs starts after the contacts that are actuated by the Initiation relay and the Manual Trip for each set of breakers. The Initiation relay actuated contacts and the upstream circuitry are considered to be RPS Logic. Manual Trip contacts and upstream circuitry are considered to be Manual Trip circuitry.

A Note associated with the ACTIONS states that if one RTCB has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected RTCB may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case the redundant RTCB will provide protection. If a single matrix power supply or vital bus failure has opened two RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.

4. Manual Trip

The LCO requires all four Manual Trip channels to be OPERABLE in MODES 1 and 2, and MODES 3, 4, and 5 when the RTCBs are closed and any CEA is capable of being withdrawn.

(continued)

BASES

LCO

4. Manual Trip (continued)

Four independent push buttons are provided. Each push button is considered a channel and operates one of the four RTCBs.

Depressing either of the two pushbuttons in both trip legs will cause an interruption of power to the CEDMs, allowing the CEAs to fall into the core. This design ensures that no single failure in any push button circuit can either cause or prevent a reactor trip.

Manual Trip push buttons are also provided at the reactor trip switchgear (locally) in case the control room push buttons become inoperable or the control room becomes uninhabitable. These are not part of the RPS and cannot be credited in fulfilling the LCO OPERABILITY requirements. Furthermore, LCO ACTIONS need not be entered due to failure of a local Manual Trip.

APPLICABILITY

This LCO is applicable to the RPS Matrix Logic, Initiation Logic, RTCB, and Manual Trips in MODES 1, 2, 3, 4, and 5. The RPS Instrumentation in MODES 1 and 2 is addressed in LCO 3.3.1. The RPS Instrumentation in MODES 3, 4, and 5 with any RTCB closed and any CEA capable of withdrawal is addressed in LCO 3.3.2. The requirement for the CEACs in MODES 1 and 2 are addressed in LCO 3.3.3.

The RPS Logic, RTCBs, and Manual Trip are required to be OPERABLE in any MODE when the CEAs are capable of being withdrawn off the bottom of the core (i.e., RTCBs closed and power available to the CEDMs). This ensures that the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with the RTCBs open, the CEAs are not capable of withdrawal and these functions do not have to be OPERABLE. The indication alarm functions required to indicate a boron dilution event are addressed in LCO 3.3.12, "Boron Dilution Alarm System (BDAS)".

(continued)

BASES

ACTIONS

A.1

Condition A applies if one Matrix Logic channel is inoperable or three Matrix Logic channels inoperable due to a common power source failure de-energizing three matrix power supplies in any applicable MODE. Loss of a single vital instrument bus will de-energize one of the two matrix power supplies in up to three matrices. This is considered a single matrix failure, providing the matrix relays associated with the failed power supplies de-energize as required. The channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered.

B.1, B.2.1, and B.2.2

Condition B applies to one Initiation Logic channel, RTCB channel, or Manual Trip channel in MODES 1 and 2, since they have the same actions. MODES 3, 4, and 5, with the RTCBs shut, are addressed in Condition C. These Required Actions require opening of the affected RTCB, or the redundant RTCB in the affected Trip Leg. This removes the need for the affected Trip Leg by performing its associated safety function. With an RTCB open, the affected Functions are in one-out-of-two logic, which meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing. Therefore, a Note has been added specifying that the RTCBs associated with one inoperable channel may be closed for up to 1 hour for the performance of an RPS CHANNEL FUNCTIONAL TEST.

(continued)

BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued)

Required Action B.1 provides for opening the RTCB associated with the inoperable Trip Leg within a Completion Time of 1 hour. This Required Action is conservative, since depressing the Manual Trip push button associated with either breaker in the other trip leg will cause a reactor trip. With this configuration, a single channel failure will not prevent a reactor trip. The allotted Completion Time is adequate for opening the affected RTCB while maintaining the risk of having it closed at an acceptable level.

Required Actions B.2.1 and B.2.2 provide for opening one of the redundant RTCB in the affected Trip leg within 1 hour and opening the affected RTCB within 48 hours. These actions allow a RTCB that fails to open to remain undisturbed for 48 hours for failure analysis, while placing the plant in a conservative condition. Opening either RTCB in the affected Trip leg ensures that opening either of the RTCBs in the other Trip leg will cause a reactor trip. This places the affected functions in one-out-of-two logic, which meets redundancy requirements. The allotted Completion Time to open one of the RTCBs in the affected Trip leg is adequate for opening the affected RTCB while maintaining the risk of having it closed at an acceptable level. The allotted action time to open the affected RTCB is adequate to preserve the failure information.

C.1

Condition C applies to the failure of one Initiation Logic channel, RTCB channel, or Manual Trip channel affecting the same trip leg in MODE 3, 4, or 5 with the RTCBs closed. The channel must be restored to OPERABLE status within 48 hours. If the inoperable channel cannot be restored to OPERABLE status within 48 hours, the affected RTCB must be opened. This removes the need for the affected channel by performing its associated safety function. With a RTCB open, the affected functions are in one-out-of-two logic, which meets redundancy requirements.

The Completion Time of 48 hours is consistent with that of other RPS instrumentation and should be adequate to repair most failures.

(continued)

BASES

ACTIONS

C.1 (continued)

Testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCB in the inoperable channels is closed to permit testing. Therefore, a Note has been added specifying that the RTCB associated with one inoperable channel may be closed for up to 1 hour for the performance of an RPS CHANNEL FUNCTIONAL TEST.

D.1

Condition D applies to the failure of both Initiation Logic channels or manual trips affecting the same trip leg. Since this will open two channels of RTCBs, this Condition is also applicable to RTCB channels in the same trip leg. This will open both sets of RTCBs in the affected trip leg, satisfying the Required Action of opening the affected RTCBs.

Of greater concern is the failure of the initiation circuit in a nontrip condition (e.g., due to two initiation relay failures). With only one Initiation Logic channel failed in a nontrip condition, there is still the redundant RTCB in the trip leg. With both failed in a nontrip condition, the reactor will not trip automatically when required. In either case the affected RTCBs must be opened immediately by using the appropriate Manual Trip push buttons, since each of the four push buttons opens one of RTCB, independent of the initiation circuitry. Caution must be exercised, since depressing the wrong push buttons may result in a reactor trip.

If the affected RTCBs cannot be opened, Required Action E is entered. This would only occur if there is a failure in the Manual Trip circuitry or the RTCB(s).

E.1 and E.2

Condition E is entered if Required Actions associated with Condition A, B, or D are not met within the required Completion Time or, if for one or more Functions, more than one Manual Trip, Matrix Logic, Initiation Logic, or RTCB channel is inoperable for reasons other than Condition A or D.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

If the RTCB associated with the inoperable channel, or the redundant RTCB in the affected Trip Leg cannot be opened, the reactor must be shut down within 6 hours and all the RTCBs opened. A Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required plant conditions from full power conditions in an orderly manner and without challenging plant systems and for opening RTCBs. All RTCBs should then be opened, placing the plant in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel and Manual Trip channel is performed every 92 days to ensure the entire channel will perform its intended function when needed.

The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 3. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. The first test, the bistable test, is addressed by SR 3.3.1.7 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS Logic: Matrix Logic and Trip Path.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected RTCB. The RTCB must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

During the Matrix Logic and Initiation Logic tests, power is applied to the Matrix relay tests coils. The test coils prevent an actuation during testing by preventing the Matrix relay contacts in the Initiation Logic from changing state during the test. This does not affect the Operability of the Initiation Logic since only one of the six logic combinations that are available to trip the Initiation Logic are affected during the test because only one Matrix Logic combination can be tested at any time. The remaining five matrix combinations available ensure that a trip in any three channels will de-energize all four Initiation paths.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

SR 3.3.4.2

Each RTCB is actuated by an undervoltage coil and a shunt trip coil. The system is designed so that either de-energizing the undervoltage coil or energizing the shunt trip coil will cause the circuit breaker to open. When an RTCB is opened, either during an automatic reactor trip or by using the manual push buttons in the control room, the undervoltage coil is de-energized and the shunt trip coil is energized. This makes it impossible to determine if one of the coils or associated circuitry is defective.

Therefore, once every 18 months, and following maintenance or adjustment of the reactor trip breakers, a CHANNEL FUNCTIONAL TEST is performed that individually tests all four undervoltage coils and all four shunt trip coils. During undervoltage coil testing, the shunt trip coils must remain de-energized, preventing their operation. Conversely, during shunt trip coil testing, the undervoltage coils must remain energized, preventing their operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2 (continued)

This Surveillance ensures that every undervoltage coil and every shunt trip coil is capable of performing its intended function and that no single active failure of any RTCB component will prevent a reactor trip. The 18 month Frequency is based on operating experience that has shown these components usually pass the Surveillance when performed at the Frequency of once every 18 months.

SR 3.3.4.3

A CHANNEL FUNCTIONAL TEST on each RTCB is performed every 31 days. This verifies proper operation of each RTCB. The RTCB must then be closed prior to testing the other three initiation circuits, or a Reactor Trip may result. The frequency of 31 days is based on the reliability analysis presented in Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A.
2. 10 CFR 100.
3. UFSAR, Section 7.2.
4. NRC Safety Evaluation Report, July 15, 1994.
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989, and Calculation 13-JC-SB-200.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and ensures acceptable consequences during accidents.

The ESFAS contains devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS);
2. Containment Spray Actuation Signal (CSAS);
3. Containment Isolation Actuation Signal (CIAS);
4. Main Steam Isolation Signal (MSIS);
5. Recirculation Actuation Signal (RAS); and
6. 7. Auxiliary Feedwater Actuation Signal (AFAS).

Equipment actuated by each of the above signals is identified in the UFSAR (Ref. 1).

Each of the above ESFAS instrumentation systems is segmented into three interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units; and
- ESFAS Logic:
  - Matrix Logic,
  - Initiation Logic (trip paths), and
  - Actuation Logic.

(continued)

BASES

BACKGROUND (continued)

This LCO addresses measurement channels and bistables. Logic is addressed in LCO 3.3.6, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip."

The role of each of these modules in the ESFAS, including the logic of LCO 3.3.6, is discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels with electrical and physical separation are provided for each parameter used in the generation of trip signals. These channels are designated A through D. Measurement channels provide input to ESFAS bistables within the same ESFAS channel. In addition, some measurement channels are used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room. Measurement channels used as an input to the RPS or ESFAS are not used for control Functions.

When a channel monitoring a parameter indicates an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistables monitoring the same parameter will de-energize Matrix Logic, which in turn de-energizes the Initiation Logic. This causes both channels of Actuation Logic to de-energize. Each channel of Actuation Logic controls one train of the associated Engineered Safety Features (ESF) equipment.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic.

(continued)

BASES

BACKGROUND

Measurement Channels (continued)

Since no single failure will either cause or prevent a protective system actuation, and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. 4).

Bistable Trip Units

Bistable trip units, mounted in the Plant Protection System (PPS) cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Matrix Logic for each ESFAS Function. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A through D, for each ESFAS Function, one for each measurement channel. In cases where two ESF Functions share the same input and trip setpoint (e.g., containment pressure input to CIAS and SIAS), the same bistable may be used to satisfy both Functions. Similarly, bistables may be shared between the RPS and ESFAS (e.g., Pressurizer Pressure–Low input to the RPS and SIAS). Bistable output relays de-energize when a trip occurs, in turn de-energizing bistable relays mounted in the PPS relay card racks.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate an ESF actuation (two-out-of-four logic).

(continued)

BASES

BACKGROUND

Bistable Trip Units (continued)

The trip setpoints and Allowable Values used in the bistables are based on the analytical limits stated in Reference 5. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment effects, for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), Allowable Values specified in Table 3.3.5-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 7). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that Safety Limits of LCO Section 2.0, "Safety Limits," are not violated during AOOs and the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Functional testing of the ESFAS, from the bistable input through the opening of initiation relay contacts in the ESFAS Actuation Logic, can be performed either at power or at shutdown and is normally performed on a quarterly basis. UFSAR, Section 7.2 (Ref. 8), provides more detail on ESFAS testing. Process transmitter calibration is normally performed on a refueling basis. SRs for the channels are specified in the Surveillance Requirements section.

(continued)

BASES

BACKGROUND (continued)

ESFAS Logic

The ESFAS Logic, consisting of Matrix, Initiation and Actuation Logic, employs a scheme that provides an ESF actuation of both trains when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic.

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one relay contact from each matrix relay in each of the four trip paths. Each trip path controls two initiation relays. Each of the two initiation relays in each trip path controls contacts in the Actuation Logic for one train of ESF.

Each of the two channels of Actuation Logic, mounted in the Auxiliary Relay Cabinet (ARCs), is responsible for actuating one train of ESF equipment. Each ESF Function has separate Actuation Logic in each ARC.

The contacts from the Initiation Logic are configured in a selective two-out-of-four logic in the Actuation Logic, similar to the configuration employed by the RPS in the RTCBs. This logic controls ARC mounted subgroup relays, which are normally energized. Contacts from these relays, when de-energized, actuate specific ESF equipment.

When a coincidence occurs in two ESFAS channels, all four matrix relays in the affected matrix will de-energize. This in turn will de-energize all eight initiation relays, four used in each Actuation Logic.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable relay cards, up to but not including the matrix relays. Matrix contacts on the bistable relay cards,

(continued)

BASES

BACKGROUND

ESFAS Logic (continued)

are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and the initiation relays.

Actuation Logic consists of all circuitry housed within the ARCs used to actuate the ESF Function, excluding the subgroup relays, and interconnecting wiring to the initiation relay contacts mounted in the PPS cabinet.

The subgroup relays are actuated by the ESFAS logic. Each ESFAS Function typically employs several subgroup relays, with each subgroup relay responsible for actuating one or more components in the ESFAS Function. Subgroup relays and their contacts are considered part of the actuated equipment and are addressed under the applicable LCO for this equipment. Initiation and Actuation Logic up to the subgroup relays is addressed in LCO 3.3.6.

It is possible to change the two-out-of-four ESFAS logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions

(continued)

BASES

BACKGROUND

ESFAS Logic (continued)

of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel.

Thus, the bistables will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

In addition to the trip channel bypasses, there are also operating bypasses on select ESFAS actuation trips. These bypasses are enabled manually in all four channels when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Operating bypasses normally are implemented in the bistable, so that normal trip indication is also disabled. The Pressurizer Pressure–Low input to the SIAS shares an operating bypass with the Pressurizer Pressure–Low reactor trip.

Manual ESFAS initiation capability is provided to permit the operator to manually actuate an ESF System when necessary.

Four handswitches (located in the control room) for each ESF Function are provided, and each handswitch actuates both trains. Each Manual Trip handswitch opens one trip path, de-energizing one set of two initiation relays, one affecting each train of ESF. Initiation relay contacts are arranged in a selective two-out-of-four configuration in the Actuation Logic. Operating either handswitch in both trip legs will result in an ESFAS Actuation. This arrangement ensures that Manual actuation will not be prevented in the event of a single random failure. Each handswitch is designated a single channel in LCO 3.3.6.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident.

An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be the secondary, or backup, actuation signal for one or more other accidents.

ESFAS protective Functions are as follows:

1. Safety Injection Actuation Signal

   SIAS ensures acceptable consequences during large break loss of coolant accidents (LOCAs), small break LOCAs, control element assembly ejection accidents, and main steam line breaks (MSLBs) inside containment. To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS) and performs several other functions such as initiating control room filtration, and starting the diesel generators.

2. Containment Spray Actuation Signal

   CSAS actuates containment spray, preventing containment overpressurization during large break LOCAs, small break LOCAs, and MSLBs or feedwater line breaks (FWLBs) inside containment. CSAS is initiated by high high containment pressure.

3. Containment Isolation Actuation Signal

   CIAS ensures acceptable mitigating actions during large and small break LOCAs, and MSLBs either inside or outside containment, and FWLBs inside containment. CIAS is initiated by low pressurizer pressure or high containment pressure.

4. Main Steam Isolation Signal

   MSIS ensures acceptable consequences during an MSLB or FWLB (between the steam generator and the main feedwater check valve), either inside or outside containment. MSIS isolates both steam generators if either generator indicates a low pressure condition, a

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

4. Main Steam Isolation Signal (continued)

high level condition or if a high containment pressure condition exists. This prevents an excessive rate of heat extraction and subsequent cooldown of the RCS during these events.

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the Refueling Water Tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level–Low signal initiates the RAS.

6, 7. Auxiliary Feedwater Actuation Signal

AFAS consists of two steam generator (SG) specific signals (AFAS-1 and AFAS-2). AFAS-1 initiates auxiliary feed to SG #1, and AFAS-2 initiates auxiliary feed to SG #2.

AFAS maintains a steam generator heat sink during a steam generator tube rupture event and an MSLB or FWLB event either inside or outside containment.

Low steam generator water level initiates auxiliary feed to the affected steam generator, providing the generator is not identified (by the rupture detection circuitry) as faulted (a steam or FWLB).

AFAS logic includes steam generator specific inputs from the SG Pressure Difference–High (SG #1 > SG #2 or SG #2 > SG #1, bistable comparators) to determine if a fault in either generator has occurred.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

6, 7. Auxiliary Feedwater Actuation Signal (continued)

Not feeding a faulted generator prevents containment overpressurization during the analyzed events.

The ESFAS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requires all channel components necessary to provide an ESFAS actuation to be OPERABLE.

The Bases for the LCOs on ESFAS Functions are:

1. Safety Injection Actuation Signal

a. Containment Pressure–High

This LCO requires four channels of Containment Pressure–High to be OPERABLE in MODES 1, 2 and 3.

The Containment Pressure–High signal is shared among the SIAS (Function 1), CIAS (Function 3), and MSIS (Function 4).

The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. The setting is low enough to initiate the ESF Functions when an abnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

b. Pressurizer Pressure–Low

This LCO requires four channels of Pressurizer Pressure–Low to be OPERABLE in MODES 1, 2 and 3.

(continued)

BASES

LCO

b. Pressurizer Pressure-Low (continued)

The Allowable Value for this trip is set low enough to prevent actuating the ESF Functions (SIAS and CIAS) during normal plant operation and pressurizer pressure transients. The setting is high enough that, with the specified accidents, the ESF systems will actuate to perform as expected, mitigating the consequences of the accident.

The Pressurizer Pressure-Low trip setpoint, which provides SIAS, CIAS, and RPS trip, may be manually decreased to a floor value of 100 psia to allow for a controlled cooldown and depressurization of the RCS without causing a reactor trip, CIAS, or SIAS. The margin between actual pressurizer pressure and the trip setpoint must be maintained less than or equal to the specified value (400 psia) to ensure a reactor trip, CIAS, and SIAS will occur if required during RCS cooldown and depressurization. When the RCS cold leg temperature is $\geq$ 485°F the setpoint must be $\geq$ 140 psia greater than the saturation pressure of the RCS cold leg. This is required to ensure a SIAS prior to reactor vessel upper head void formation in the event of RCS depressurization caused by a steam line break.

From this reduced setting, the trip setpoint will increase automatically as pressurizer pressure increases, tracking actual RCS pressure until the trip setpoint is reached.

When the trip setpoint has been lowered below the bypass permissive setpoint of 400 psia, the Pressurizer Pressure-Low reactor trip, CIAS, and SIAS actuation may be manually bypassed in preparation for shutdown cooling. When RCS pressure rises above the bypass removal setpoint, the bypass is removed.

(continued)

BASES

LCO
(continued)

Bypass Removal

This LCO requires four channels of operating bypass removal for Pressurizer Pressure–Low to be OPERABLE in MODES 1, 2 and 3.

Each of the four channels enables and disables the operating bypass capability for a single channel. Therefore, this LCO applies to the operating bypass removal feature only. If the bypass enable function is failed so as to prevent entering a operating bypass condition, operation may continue.

Because the trip setpoint has a floor value of 100 psia, a channel trip will result if pressure is decreased below this setpoint without bypassing.

The operating bypass removal Allowable Value was chosen because MSLB events originating from below this setpoint add less positive reactivity than that which can be compensated for by required SDM.

2. Containment Spray Actuation Signal

a. Containment Pressure–High High

This LCO requires four channels of Containment Pressure–High High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e. plant heatup) and is not indicative of an abnormal condition. The setting is low enough to initiate CSAS in time to prevent containment pressure from exceeding design.

(continued)

BASES

LCO (continued)

3. Containment Isolation Actuation Signal

The SIAS and CIAS are actuated on Pressurizer Pressure–Low or Containment Pressure–High, the SIAS and CIAS share the same input channels, bistables, and matrices and matrix relays. The remainder of the initiation channels, the manual channels, and the Actuation Logic are separate and are addressed in LCO 3.3.6.

a. Containment Pressure–High

This LCO requires four channels of Containment Pressure–High to be OPERABLE in MODES 1, 2, and 3.

The Containment Pressure–High signal is shared among the SIAS (Function 1), CIAS (Function 3), and MSIS (Function 4).

The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. The setting is low enough to initiate the ESF Functions when an abnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

b. Pressurizer Pressure–Low

This LCO requires four channels of Pressurizer Pressure–Low to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is set low enough to prevent actuating the ESF Functions (SIAS and CIAS) during normal plant operation and pressurizer pressure transients. The setting is high enough that, with the specified accident, the ESF systems will actuate to perform as expected, mitigating the consequences of the accidents.

(continued)

BASES

LCO

b. Pressurizer Pressure-Low (continued)

The Pressurizer Pressure-Low trip setpoint, which provides an SIAS, CIAS, and RPS trip, may be manually decreased to a floor Allowable Value of 100 psia to allow for a controlled cooldown and depressurization of the RCS without causing a reactor trip, CIAS or SIAS.

The safety margin between actual pressurizer pressure and the trip setpoint must be maintained less than or equal to the specified value (400 psi) to ensure a reactor trip, CIAS, and SIAS will occur if required during RCS cooldown and depressurization.

From this reduced setting, the trip setpoint will increase automatically as pressurizer pressure increases, tracking actual RCS pressure until the trip setpoint is reached.

When the trip setpoint has been lowered below the operating bypass permissive setpoint of 400 psia, the Pressurizer Pressure-Low reactor trip, CIAS, and SIAS actuation may be manually bypassed in preparation for shutdown cooling. When RCS pressure rises above the bypass removal, the bypass is removed.

Bypass Removal

This LCO requires four channels of operating bypass removal for Pressurizer Pressure-Low to be OPERABLE in MODES 1, 2, and 3.

Each of the four channels enables and disables the operating bypass capability for a single channel. Therefore all four operating bypass removal channels must be OPERABLE to ensure that none of the four channels are inadvertently bypassed.

(continued)

BASES

LCO

Bypass Removal (continued)

This LCO applies to the operating bypass removal feature only. If the operating bypass enable function is failed so as to prevent entering a operating bypass condition, operation may continue. Because the trip setpoint has a floor value of 100 psia, a channel trip will result if pressure is decreased below this setpoint without bypassing.

The operating bypass removal Allowable Value was chosen because MSLB events originating from below this setpoint add less positive reactivity than that which can be compensated for by required SDM.

4. Main Steam Isolation Signal

The LCO is applicable to the MSIS in MODES 1, 2 and 3 except when all associated valves are closed.

a. Steam Generator Pressure - Low

This LCO requires four channels of Steam Generator Pressure - Low to be OPERABLE in MODES 1, 2 and 3.

The Allowable Value for this trip is set below the full load operating value for steam pressure so as not to interfere with normal plant operation. However, the setting is high enough to provide an MSIS (Function 4) during an excessive steam demand event. An excessive steam demand event causes the RCS to cool down, resulting in a positive reactivity addition to the core.

MSIS limits this cooldown by isolating both steam generators if the pressure in either drops below the trip setpoint. An RPS trip on Steam Generator Pressure - Low is initiated simultaneously, using the same bistable.

(continued)

BASES

LCO

a. Steam Generator Pressure - Low (continued)

The Steam Generator Pressure–Low trip setpoint may be manually decreased as steam generator pressure is reduced. This prevents an RPS trip or MSIS actuation during controlled plant cooldown. The margin between actual steam generator pressure and the trip setpoint must be maintained less than or equal to the specified value of 200 psia to ensure a reactor trip and MSIS will occur when required.

b. Containment Pressure–High

This LCO requires four channels of Containment Pressure–High to be OPERABLE in MODES 1, 2 and 3. The Containment Pressure–High signal is shared among the SIAS (Function 1), CIAS (Function 3), and MSIS (Function 4).

The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. The setting is low enough to initiate the ESF Functions when an abnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

c. Steam Generator Level-High

This LCO requires four channels of Steam Generator Level-High to be OPERABLE in MODES 1, 2 and 3.

The allowable value for this trip is set high enough to ensure it does not interfere with normal plant operation. The setting is low enough to prevent moisture damage to secondary plant components in the case of a steam generator overfill event.

(continued)

BASES

LCO (continued)

5. Recirculation Actuation Signal

a. Refueling Water Tank Level – Low

This LCO requires four channels of RWT Level – Low to be OPERABLE in MODES 1, 2, and 3.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump.

Premature recirculation could impair the reactivity control function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water to prevent air entrainment in the suction.

The lower limit on the RWT Level – Low trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the RWT.

6, 7. Auxiliary Feedwater Actuation Signal SG #1 and SG #2 (AFAS-1 and AFAS-2)

AFAS-1 is initiated to SG #1 by either a low steam generator level coincident with no differential pressure trip present or by a low steam generator level coincident with a differential pressure between the two generators with the higher pressure in SG #1. AFAS-2 is similarly configured to feed SG #2.

The steam generator secondary differential pressure is used, as an input of the AFAS logic where it is used to determine if a generator is intact. The AFAS logic inhibits feeding a steam generator if the pressure in that steam generator is less than the pressure in the other steam generator by the Steam Generator Pressure Difference (SGPD) – High setpoint.

(continued)

BASES

LCO

6. 7. Auxiliary Feedwater Actuation Signal SG #1 and SG #2 (AFAS-1 and AFAS-2) (continued)

The SGPD setpoint is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation.

The following LCO description applies to both AFAS signals.

a. Steam Generator Level-Low

This LCO requires four channels of Steam Generator Level-Low to be OPERABLE for each AFAS in MODES 1, 2, and 3.

The Steam Generator Level-Low AFAS input is shared with the Steam Generator Level-Low RPS function. The Steam Generator Level-Low AFAS and RPS use separate bistables. This allows the AFAS setpoint to be set lower than the RPS setpoint. The allowable value is high enough to ensure the steam generator is available as a heat sink. The setting is low enough to prevent inadvertent AFAS actuations during plant transients. This setpoint provides allowance that there will be sufficient inventory in the steam generator at the time of the RPS trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

b. SG Pressure Difference-High (SG #1 > SG #2) or (SG #2 > SG #1)

This LCO requires four channels of SG Pressure Difference-High to be OPERABLE for each AFAS in MODES 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation without an actuation. The setting is low enough

(continued)

BASES

LCO

b. SG Pressure Difference – High (SG #1 > SG #2) or (SG #2 > SG #1) (continued)

to detect and inhibit feeding of a faulted (MSLB or FWLB) steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

APPLICABILITY

In MODES 1, 2 and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition;
- Actuate auxiliary feedwater to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and
- Actuate ESF systems to ensure sufficient borated water inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 4, 5 and 6 automatic actuation of these Functions is not required because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required, as addressed by LCO 3.3.6.

Several trips have operating bypasses, discussed in the preceding LCO section. The interlocks that allow these bypasses shall be OPERABLE whenever the RPS Function they support is OPERABLE.

(continued)

BASES

ACTIONS

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection Function affected.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass on trip all associated functional units as listed below.

Process Measurement Circuit

1.	Steam Generator Pressure-Low	Steam Generator Pressure-Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2.	Steam Generator Level (Wide Range)	Steam Generator Level-Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

With a Steam Generator Pressure Difference-High channel inoperable or in test, bypass or trip the associated Steam Generator Level-Low (ESF) function.

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately, if applicable in the current MODE of operation.

A Note has been added to the ACTIONS. The Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time for the inoperable channel of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

Condition A applies to the failure of a single channel of one or more input parameters in the following ESFAS Functions:

1. Safety Injection Actuation Signal
   Containment Pressure – High
   Pressurizer Pressure – Low

2. Containment Spray Actuation Signal
   Containment Pressure – High High

3. Containment Isolation Actuation Signal
   Containment Pressure – High
   Pressurizer Pressure – Low

4. Main Steam Isolation Signal
   Steam Generator #1 Pressure – Low
   Steam Generator #2 Pressure – Low
   Steam Generator #1 Level-High
   Steam Generator #2 Level-High
   Containment Pressure – High

5. Recirculation Actuation Signal
   Refueling Water Storage Tank Level – Low

6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)
   Steam Generator #1 Level – Low
   SG Pressure Difference (SG #2 > SG #1) – High

7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)
   Steam Generator #2 Level – Low
   SG Pressure Difference (SG #1 > SG #2) – High

ESFAS coincidence logic is normally two-out-of-four.

If one ESFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action A.1).

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of 1 hour allotted to restore, bypass, or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

The failed channel must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. With a channel bypassed, the coincidence logic is now in a two-out-of-three configuration. The Completion Time of prior to entering MODE 2 following the next MODE 5 entry is based on adequate channel to channel independence, which allows a two-out-of-three channel operation, since no single failure will cause or prevent an ESF actuation.

B.1

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

Condition B applies to the failure of two channels of one or more input parameters in the following ESFAS automatic trip Functions:

1. Safety Injection Actuation Signal
   Containment Pressure–High
   Pressurizer Pressure–Low

2. Containment Spray Actuation Signal
   Containment Pressure–High High

3. Containment Isolation Actuation Signal
   Containment Pressure–High
   Pressurizer Pressure–Low

(continued)

BASES

ACTIONS

B.1 (continued)

4. Main Steam Isolation Signal
   Steam Generator #1 Pressure – Low
   Steam Generator #2 Pressure – Low
   Steam Generator #1 Level-High
   Steam Generator #2 Level-High
   Containment Pressure-High

5. Recirculation Actuation Signal
   Refueling Water Storage Tank Level – Low

6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)
   Steam Generator #1 Level – Low
   SG Pressure Difference (SG #2 > SG #1) – High

7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)
   Steam Generator #2 Level – Low
   SG Pressure Difference (SG #1 > SG #2) – High

With two inoperable channels, power operation may continue, provided one inoperable channel is placed in bypass and the other channel is placed in trip within 1 hour. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic in the bypassed input parameter, but with another channel failed, the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the two inoperable channels will need to be restored to OPERABLE status prior to the next required CHANNEL FUNCTIONAL TEST because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not possible to bypass more than one ESFAS channel, and placing a second channel in trip will result in an ESFAS actuation. Therefore, if one ESFAS channel is in trip and a second channel is in bypass, a third inoperable channel would place the unit in LCO 3.0.3.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1, and C.2.2

Condition C applies to one automatic operating bypass removal channel inoperable. The only automatic operating bypass removal on an ESFAS is on the Pressurizer Pressure–Low signal. This operating bypass removal is shared with the RPS Pressurizer Pressure–Low bypass removal.

If the bypass removal channel for any operating bypass cannot be restored to OPERABLE status, the associated ESFAS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected ESFAS channel must be declared inoperable, as in Condition A, and the operating bypass either removed or the bypass removal channel repaired. The Bases for the Required Actions and required Completion Times are consistent with Condition A.

D.1 and D.2

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

Condition D applies to two inoperable automatic operating bypass removal channels. If the operating bypass removal channels for two operating bypasses cannot be restored to OPERABLE status, the associated ESFAS channel may be considered OPERABLE only if the operating bypass is not in effect. Otherwise, the affected ESFAS channels must be declared inoperable, as in Condition B, and either the operating bypass removed or the bypass removal channel repaired. The restoration of one affected bypassed automatic trip channel must be completed prior to the next CHANNEL FUNCTIONAL TEST or the plant must shut down per LCO 3.0.3, as explained in Condition B. Completion Times are consistent with Condition B.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, C, or D cannot be met, 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.

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1 (continued)

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed.

The CHANNEL FUNCTIONAL TEST is part of an overlapping test sequence similar to that employed in the RPS. This sequence, consisting of SR 3.3.5.2, SR 3.3.6.1, and SR 3.3.6.2, tests the entire ESFAS from the bistable input through the actuation of the individual subgroup relays. These overlapping tests are described in Reference 1. SR 3.3.5.2 and SR 3.3.6.1 are normally performed together and in conjunction with ESFAS testing. SR 3.3.6.2 verifies that the subgroup relays are capable of actuating their respective ESF components when de-energized.

These tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components. SRs 3.3.6.1 and 3.3.6.2 are addressed in LCO 3.3.6. SR 3.3.5.2 includes bistable tests.

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. This is done with the affected RPS trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2 (continued)

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 9.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector and the bypass removal functions. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 9.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

SR 3.3.5.4

This Surveillance ensures that the train actuation response times are within the maximum values assumed in the safety analyses.

Response time testing acceptance criteria are included in Reference 8.

ESF RESPONSE TIME tests are conducted on a STAGGERED TEST BASIS of once every 18 months. The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.5

SR 3.3.5.5 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.5.2, except SR 3.3.5.5 is performed within 92 days prior to startup and is only applicable to operating bypass functions. Since the Pressurizer Pressure–Low operating bypass is identical for both the RPS and ESFAS, this is the same Surveillance performed for the RPS in SR 3.3.1.13.

The CHANNEL FUNCTIONAL TEST for proper operation of the operating bypass permissives is critical during plant heatups because the bypasses may be in place prior to entering MODE 3 but must be removed at the appropriate points during plant startup to enable the ESFAS Function. Consequently, just prior to startup is the appropriate time to verify operating bypass function OPERABILITY. Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated ESFAS Function is inappropriately bypassed. This feature is verified by SR 3.3.5.2.

The allowance to conduct this test with 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 9).

REFERENCES

1. UFSAR, Section 7.3.
2. 10 CFR 50, Appendix A.
3. NRC Safety Evaluation Report, July 15, 1994
4. IEEE Standard 279-1971.
5. UFSAR, Chapter 15.
6. 10 CFR 50.49.
7. "Calculation of Trip Setpoint Valves Plant Protection System", CEN-286(v), or Calculation 13-JC-SG-203 for the Low Steam Generator Pressure Trip Function.
8. UFSAR, Section 7.2.
9. CEN-327, May 1986, including Supplement 1, March 1989, and Calculation 13-JC-SB-200.

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and ensures acceptable consequences during accidents.

The ESFAS contains devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS);
2. Containment Isolation Actuation Signal (CIAS);
3. Recirculation Actuation Signal (RAS);
4. Containment Spray Actuation Signal (CSAS);
5. Main Steam Isolation Signal (MSIS);
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1); and
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2).

Equipment actuated by each of the above signals is identified in the UFSAR (Ref. 1).

Each of the above ESFAS instrumentation systems is segmented into three interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units; and
- ESFAS Logic:

(continued)

BASES

BACKGROUND (continued)

- Matrix Logic,
- Initiation Logic (trip paths), and
- Actuation Logic.

This LCO addresses ESFAS Logic. Bistables and measurement channels are addressed in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation."

The role of the measurement channels and bistables is described in LCO 3.3.5. The role of the ESFAS Logic is described below.

ESFAS Logic

The ESFAS Logic, consisting of Matrix, Initiation and Actuation Logic, employs a scheme that provides an ESF actuation of both trains when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic.

Bistable relay contact outputs from the four channels are configured into six Matrix Logics. Each Matrix Logic checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices, to reflect the bistable channels being monitored. Each Matrix Logic contains four normally energized matrix relays. When a coincidence is detected in the two channels being monitored by the Matrix Logic, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one relay contact from each matrix relay in each of the four trip paths. Each trip path controls two initiation relays. Each of the two initiation relays in each trip path controls contacts in the Actuation Logic for one train of ESF.

Each of the two channels of Actuation Logic, mounted in the Auxiliary Relay Cabinets (ARCs), is responsible for actuating one train of ESF equipment. Each ESF Function has separate Actuation Logic in each ARC.

The contacts from the Initiation Logic are configured in a selective two-out-of-four logic in the Actuation Logic, similar to the configuration employed by the RPS in the RTCBs. This logic controls ARC mounted subgroup relays,

(continued)

BASES

BACKGROUND

ESFAS Logic (continued)

which are normally energized. Contacts from these relays, when de-energized, actuate specific ESF equipment.

When a coincidence occurs in two ESFAS channels, all four matrix relays in the affected matrix will de-energize. This, in turn, will de-energize all eight initiation relays, four used in each Actuation Logic.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts and interconnecting matrix wiring between bistable relay cards, up to but not including the matrix relays. Matrix contacts on the bistable relay cards are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and the initiation relays.

Actuation Logic consists of all circuitry housed within the ARCs used to actuate the ESF Function, excluding the subgroup relays, and interconnecting wiring to the initiation relay contacts mounted in the PPS cabinet.

(continued)

BASES

BACKGROUND

ESFAS Logic (continued)

The subgroup relays are actuated by the ESFAS Logic. Each ESFAS Function typically employs several subgroup relays, with each subgroup relay responsible for actuating one or more components in the ESFAS Function. Subgroup relays and their contacts are considered part of the actuated equipment and are addressed under the applicable LCO for this equipment.

It is possible to change the two-out-of-four ESFAS Logic to two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing. Trip channel bypassing is addressed in LCO 3.3.5.

Manual ESFAS initiation capability is provided to permit the operator to manually actuate an ESF System when necessary.

Four handswitches (located in the control room) for each ESF Function are provided, and each handswitch actuates both trains. Each Manual Trip handswitch opens one trip path, de-energizing one set of two initiation relays, one affecting each train of ESF. Initiation relay contacts are arranged in a selective two-out-of-four configuration in the Actuation Logic. Operating either handswitch in both Trip Legs will result in an ESFAS Actuation. This arrangement ensures that Manual Actuation will not be prevented in the event of a single random failure. Each handswitch is designated a single channel in this LCO.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents.

ESFAS Functions are as follows:

1. Safety Injection Actuation Signal

   SIAS ensures acceptable consequences during large break loss of coolant accidents (LOCAs), small break LOCAs, control element assembly ejection accidents, and main steam line breaks (MSLBs) inside containment. To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS) and performs several other Functions, such as initiating control room isolation, and starting the diesel generators.

2. Containment Isolation Actuation Signal

   CIAS ensures acceptable mitigating actions during large and small break LOCAs and during MSLBs either inside or outside containment and feedwater line breaks (FWLBs) inside containment. CIAS is initiated by low pressurizer pressure or high containment pressure.

3. Recirculation Actuation Signal

   At the end of the injection phase of a LOCA, the Refueling Water Tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

3. Recirculation Actuation Signal (continued)

Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level–Low signal initiates the RAS.

4. Containment Spray Actuation Signal

CSAS actuates containment spray, preventing containment overpressurization during large break LOCAs, small break LOCAs, and MSLBs or FWLBs inside containment. CSAS is initiated by high high containment pressure.

5. Main Steam Isolation Signal

MSIS ensures acceptable consequences during an MSLB or FWLB (between the steam generator and the main feedwater check valve) either inside or outside containment. MSIS isolates both steam generators if either generator indicates a low pressure condition or a high level condition or if a high containment pressure condition exists. This prevents an excessive rate of heat extraction and subsequent cooldown of the RCS during these events.

6, 7. Auxiliary Feedwater Actuation Signal

AFAS consists of two Steam Generator (SG) specific signals AFAS-1 and AFAS-2. AFAS-1 initiates auxiliary feed to SG #1, and AFAS-2 initiates auxiliary feed to SG #2.

AFAS maintains a steam generator heat sink during a steam generator tube rupture event and an MSLB or FWLB event either inside or outside containment.

Low steam generator water level initiates auxiliary feed to the affected steam generator, providing the generator is not identified (by the rupture detection circuitry) as faulted (an MSLB or FWLB).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

6, 7. Auxiliary Feedwater Actuation Signal (continued)

AFAS logic includes steam generator specific inputs from the SG Pressure Difference–High (SG #1 > SG #2 or SG #2 > SG #1, bistable comparators) to determine if a fault in either generator has occurred.

Not feeding a faulted generator prevents containment overpressurization during the analyzed events.

The ESFAS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO on the ESFAS Logic channels ensures that each of the following requirements are met:

- An ESFAS Actuation Signal will be initiated when necessary;
- The required protection system coincidence logic is maintained (minimum two-out-of-three, normal two-out-of-four); and
- Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

Failures of individual bistable relays and their contacts are addressed in LCO 3.3.5. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

A Matrix Logis is considered inoperable if a coincident trip in the same Function in the two OPERABLE channels monitored by the Matrix Logic will not remove power from the coils of all four Matrix relays. The OPERABILITY of the Matrix Logic is not affected by inoperable measurement channels.

Loss of a single vital bus will de-energize one of the two power supplies in each of the three matrices. This will result in two trip path contacts opening in each ESFAS Actuation Logic channel; however, the remaining two contacts in each ESFAS Actuation Logic channel will remain

(continued)

BASES

closed, preventing an ESFAS Actuation. For the purposes of this LCO, de-energizing up to three matrix power supplies due to a single failure is to be treated as a single channel failure, providing the affected matrix relays de-energize as designed, opening the affected trip path contacts in each ESFAS Actuation Logic channel.

Each of the four Initiation Logic channels controls two Initiation relays, each Initiation relay opens a contact in its Actuation Logic channel if any of the six coincidence matricies de-energize their associated matrix relays. They thus form a logical OR function. Each Initiation Logic channel has its own power supply and is independent of the others. An Initiation Logic channel includes the matrix relay through to the Initiation relay contacts, and the interconnecting wiring to the Actuation Logic channels.

An Initiation Logic is considered inoperable if the contacts on both Initiation relays will not operate when power is removed from the coils of any of the six matrix relays in the trip path.

It is possible for two Initiation Logic channels affecting the same trip leg to de-energize if a matrix power supply or vital instrument bus fails. This will result in opening two contacts in each of the ESFAS Actuation Logic channels.

An Actuation Logic channel is inoperable if a selective two-out-of-four trip signal is received from the Initiation Logic for any ESFAS Function, and power is not removed from the coils of all of the subgroup relays actuated by that function.

The requirements for each Function are listed below. The reasons for the applicable MODES for each Function are addressed under APPLICABILITY.

1. Safety Injection Actuation Signal

   Automatic SIAS occurs in Pressurizer Pressure–Low or Containment Pressure–High and is explained in Bases 3.3.5.

   a. Manual Trip
      This LCO requires four channels of SIAS Manual Trip to be OPERABLE in MODES 1, 2, 3, and 4.

(continued)

BASES

LCO (continued)

b. Matrix Logic

This LCO requires six channels of SIAS Matrix Logic to be OPERABLE in MODES 1, 2 and 3.

c. Initiation Logic

This LCO requires four channels of SIAS Initiation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

d. Actuation Logic

This LCO requires two channels of SIAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

2. Containment Isolation Actuation Signal

The SIAS and CIAS are actuated on Pressurizer Pressure-Low or Containment Pressure-High, the SIAS and CIAS share the same input channels, bistables, and matrices and matrix relays. The remainder of the initiation channels, the manual channels, and the Actuation Logic are separate. Since their applicability is also the same, they have identical actions.

a. Manual Trip

This LCO requires four channels of CIAS Manual Trip to be OPERABLE in MODES 1, 2, 3, and 4.

b. Matrix Logic

This LCO requires six channels of CIAS Matrix Logic to be OPERABLE in MODES 1, 2, and 3.

c. Initiation Logic

This LCO requires four channels of CIAS Initiation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

d. Actuation Logic

This LCO requires two channels of CIAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

(continued)

BASES

LCO (continued)

3. Recirculation Actuation Signal

a. Manual Trip

This LCO requires four channels of RAS Manual Trip to be OPERABLE in MODES 1, 2, 3, and 4.

b. Matrix Logic

This LCO requires six channels of RAS Matrix Logic to be OPERABLE in MODES 1, 2, and 3.

c. Initiation Logic

This LCO requires four channels of RAS Initiation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

d. Actuation Logic

This LCO requires two channels of RAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

4. Containment Spray Actuation Signal

a. Manual Trip

This LCO requires four channels of CSAS Manual Trip to be OPERABLE in MODES 1, 2, and 3.

b. Matrix Logic

This LCO requires six channels of CSAS Matrix Logic to be OPERABLE in MODES 1, 2, and 3.

c. Initiation Logic

This LCO requires four channels of CSAS Initiation Logic to be OPERABLE in MODES 1, 2, and 3.

d. Actuation Logic

This LCO requires two channels of CSAS Actuation Logic to be OPERABLE in MODES 1, 2, and 3.

(continued)

BASES

LCO
(continued)

5. Main Steam Isolation Signal

a. Manual Trip

This LCO requires four channels of MSIS Manual Trip to be OPERABLE in MODES 1, 2 and 3, except when all associated valves are closed.

b. Matrix Logic

This LCO requires six channels of MSIS Matrix Logic to be OPERABLE in MODES 1, 2 and 3, except when all associated valves are closed.

c. Initiation Logic

This LCO requires four channels of MSIS Initiation Logic to be OPERABLE in MODES 1, 2 and 3, except when all associated valves are closed.

d. Actuation Logic

This LCO requires two channels of MSIS Actuation Logic to be OPERABLE in MODES 1, 2 and 3, except when all associated valves are closed.

6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)

AFAS-1 is initiated either by a low steam generator level coincident with no differential pressure trip present or by a low steam generator level coincident with a differential pressure between the two generators with the higher pressure in SG #1.

The steam generator secondary differential pressure is used, as an input of the AFAS logic where it is used to determine if a generator is intact. The AFAS logic inhibits feeding a steam generator if the pressure in that steam generator is less than the Steam Generator Pressure Difference (SGPD)–High setpoint pressure.

(continued)

BASES

LCO

6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1) (continued)

The setpoint is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation.

a. Manual Trip

This LCO requires four channels of Manual Trip to be OPERABLE in MODES 1, 2, and 3.

b. Matrix Logic

This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1, 2, and 3.

c. Initiation Logic

This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1, 2, and 3.

d. Actuation Logic

This LCO requires two channels of Actuation Logic to be OPERABLE in MODES 1, 2, and 3.

7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)

AFAS-2 is initiated either by a low steam generator level coincident with no differential pressure trip present or by a low steam generator level coincident with a differential pressure between the two generators with the higher pressure in SG #2.

The steam generator secondary differential pressure is used, as an input of the AFAS Logic where it is used to determine if a generator is intact. The AFAS Logic inhibits feeding a steam generator if the pressure in that steam generator is less than the SGPD-High setpoint pressure.

(continued)

BASES

LCO (continued)

7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)

The setpoint is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation.

a. Manual Trip

This LCO requires four channels of Manual Trip to be OPERABLE in MODES 1, 2, and 3.

b. Matrix Logic

This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1, 2, and 3.

c. Initiation Logic

This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1, 2, and 3.

d. Actuation Logic

This LCO requires two channels of Actuation Logic to be OPERABLE in MODES 1, 2, and 3.

APPLICABILITY

In MODES 1, 2 and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition;
- Actuate auxiliary feedwater to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and

(continued)

BASES

APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated water inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 4, 5, and 6, automatic actuation of these Functions is not required because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required.

ESFAS Manual Trip capability is required in MODE 4 for SIAS, CIAS, and RAS even though automatic actuation is not required. Because of the large number of components actuated by these Functions, ESFAS actuation is simplified by the use of the Manual Trip.

CSAS, MSIS, and AFAS have relatively few components, which can be actuated individually if required in MODE 4, and the systems may be disabled or reconfigured, making system level Manual Trip impossible and unnecessary.

The ESFAS logic must be OPERABLE in the same MODES as the automatic and Manual Trip. In MODE 4, only the portion of the ESFAS logic responsible for the required Manual Trip must be OPERABLE.

In MODES 5 and 6, the systems initiated by ESFAS are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

(continued)

BASES

ACTIONS

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately, if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time for the inoperable channel of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

Condition A applies if one Matrix Logic channel is inoperable. Since matrix power supplies in a given matrix (e.g., AB, BC, etc.) are common to all ESFAS Functions, a single power supply failure may affect more than one matrix.

Failures of individual bistables, their relays, and the trip channel bypass relays and their contacts are considered measurement channel failures. This section describes failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies. Loss of a single vital bus will de-energize one of the two power supplies in each of three matrices. This will result in two initiation circuits de-energizing, reducing the ESFAS Actuation Logic to a one-out-of-two logic in both trains.

Condition A also applies when de-energizing up to three matrix power supplies due to a single failure, such as loss of a vital instrument bus. This is to be treated as a single matrix channel failure, providing the affected matrix relays de-energize as designed. Although each of the six matrices within an ESFAS Function uses separate power supplies, the matrices for the different ESFAS Functions share power supplies. Thus, failure of a matrix power supply may force entry into the Condition specified for each of the affected ESFAS Functions.

(continued)

BASES

ACTIONS

A.1 (continued)

The channel must be restored to OPERABLE status within 48 hours. This provides the operator with time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour period. If the channel cannot be restored to OPERABLE status with 48 hours, Condition E is entered.

B.1

Condition B applies to one Manual Trip or Initiation Logic channel inoperable.

The channel must be restored to OPERABLE status within 48 hours. Operating experience has demonstrated that the probability of a random failure in a second channel is low during any given 48 hour period.

Failure of a single Initiation Logic channel may open one contact affecting both Actuation Logic channels. For the purposes of this Specification, the Actuation Logic is not inoperable. This prevents the need to enter LCO 3.0.3 in the event of an Initiation Logic channel failure. The Actions differ from those involving one RPS manual channel inoperable, because in the case of the RPS, opening RTCBs can be easily performed and verified. Opening an initiation relay contact is more difficult to verify, and subsequent shorting of the contact is always possible.

C.1 and C.2

Condition C applies to the failure of both Initiation Logic channels affecting the same trip leg.

In this case, the Actuation Logic channels are not inoperable, since they are in one-out-of-two logic and capable of performing as required. This obviates the need to enter LCO 3.0.3 in the event of a matrix or vital bus power failure.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Both Initiation Logic channels in the same trip leg will de-energize if a matrix power supply or vital instrument bus is lost. This will open the Actuation Logic contacts, satisfying the Required Action to open at least one set of contacts in the affected trip leg. Indefinite operation in this condition is prohibited because of the difficulty of ensuring the contacts remain open under all conditions. Thus, the channel must be restored to OPERABLE status within 48 hours. This provides the operator with time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second channel is low during any given 48 hour period. If the channel cannot be restored to OPERABLE status with 48 hours, Condition E is entered.

Of greater concern is the failure of the initiation circuit in a nontrip condition, e.g., due to two initiation relay failures. With one failed, there is still the redundant contact in the trip leg of each Actuation Logic. With both failed in a nontrip condition, the ESFAS Function is lost in the affected train. To prevent this, immediate opening of at least one contact in the affected trip leg is required. If the required contact has not opened, as indicated by annunciation or trip leg current lamps, Manual Trip of the affected trip leg contacts may be attempted. Caution must be exercised, since operating the wrong ESFAS handswitch may result in an ESFAS actuation.

D.1

Condition D applies to Actuation Logic.

With one Actuation Logic channel inoperable, automatic actuation of one train of ESF may be inhibited. The remaining train provides adequate protection in the event of Design Basis Accidents, but the single failure criterion may be violated. For this reason operation in this condition is restricted.

(continued)

BASES

ACTIONS

D.1 (continued)

The channel must be restored to OPERABLE status within 48 hours. Operating experience has demonstrated that the probability of a random failure in the Actuation Logic of the second train is low during a given 48 hour period.

Failure of a single Initiation Logic channel, matrix channel power supply, or vital instrument bus may open one or both contacts in the same trip leg in both Actuation Logic channels. For the purposes of this Specification, the Actuation Logic is not inoperable. This obviates the need to enter LCO 3.0.3 in the event of a vital bus, matrix, or initiation channel failure.

Each Actuation Logic channel has two sets of redundant power supplies. The power supplies in each set are powered from different vital instrument buses. Failure of a single power supply or a set of power supplies due to the loss of a vital instrument bus, does not affect the operation of the Actuation Logic because the redundant power supplies can supply the full system load. For the purposes of this specification, the Actuation Logic is not inoperable.

Required Action D.1 is modified by a Note to indicate that one channel of Actuation Logic may be bypassed for up to 1 hour for Surveillance, provided the other channel is OPERABLE.

This allows performance of a PPS CHANNEL FUNCTIONAL TEST on an OPERABLE ESFAS train without generating an ESFAS actuation in the inoperable train.

E.1 and E.2

If the Required Actions and associated Completion Times of Conditions for CSAS, MSIS or AFAS cannot be met, 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.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

If the Required Actions and associated Completion Times for SIAS, CIAS, or RAS are not met, 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. If the Required Actions and associated Completion Times for SIAS, CIAS, or RAS Matrix Logic are not met this Action may be exited when the plant is brought to MODE 4 since the LCO does not apply in MODE 4. 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 3.3.6.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed.

The CHANNEL FUNCTIONAL TEST is part of an overlapping test sequence similar to that employed in the RPS. This sequence, consisting of SR 3.3.5.2, SR 3.3.6.1, and SR 3.3.6.2, tests the entire ESFAS from the bistable input through the actuation of the individual subgroup relays. These overlapping tests are described in Reference 1. SR 3.3.5.2 and SR 3.3.6.1 are normally performed together and in conjunction with ESFAS testing. SR 3.3.6.2 verifies that the subgroup relays are capable of actuating their respective ESF components when de-energized.

These tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components. SR 3.3.5.2 is addressed in LCO 3.3.5. SR 3.3.6.1 includes Matrix Logic tests and trip path (Initiation Logic) tests, and Manual Actuation Tests.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each function removes power to the matrix relays. During testing, power is applied to the matrix relay test coils, preventing the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts.

Trip Path (Initiation Logic) Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening one contact in each Actuation Logic channel.

The initiation circuit lockout relay must be reset (except for AFAS, which lacks initiation circuit lockout relays) prior to testing the other three initiation circuits, or an ESFAS actuation may result.

Automatic Actuation Logic operation is verified during Initiation Logic testing by verifying that current is interrupted in each trip leg in the selective two-out-of-four actuation circuit logic whenever the initiation relay is de-energized. A Note is added to indicate that testing of Actuation Logic shall include verification of the proper operation of each initiation relay.

(continued)

BASES

SURVEILLANCE

Trip Path (Initiation Logic) Tests (continued)

During the Matrix Logic and Initiation Logic test, power is applied to the Matrix relay test coils. The test coils prevent an actuation during testing by preventing the Matrix relay contacts in the Initiation Logic from changing state during the test. This does not affect the Operability of the Initiation Logic since only one of the six logic combinations that are available to trip the Initiation Logic are affected during the test because only one Matrix Logic combination can be tested at any time. The remaining five matrix combinations available ensure that a trip in any three channels will de-energize all four Initiation paths.

Manual Trip Tests

This test verifies that the manual trip handswitches are capable of opening contacts in the Actuation Logic as designed.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 2).

SR 3.3.6.2

Individual ESFAS subgroup relays must also be tested, one at a time, to verify the individual ESFAS components will actuate when required. Proper operation of the individual subgroup relays is verified by de-energizing these relays one at a time using an ARC mounted test circuit. Proper operation of each component actuated by the individual relays is thus verified without the need to actuate the entire ESFAS function.

The 9 months Staggered Test Frequency is based on operating experience and ensures individual relay problems can be detected within this time frame. Considering the large number of similar relays in the ARC, and the similarity in their use, a large test sample can be assembled to verify the validity of this Frequency. The actual justification is based on CEN-403, "ESFAS Subgroup Relay Test Interval Extension (Ref. 3).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2 (continued)

Some components cannot be tested at power since their actuation might lead to plant trip or equipment damage. Reference 1 lists those relays exempt from testing at power, with an explanation of the reason for each exception. Relays not tested at power must be tested in accordance with the Note to this SR.

REFERENCES

1. UFSAR, Section 7.3.

2. CEN-327, May 1986, including Supplement 1, March 1989, and Calculation 13-JC-SB-200.

3. CEN-403, "ESFAS Subgroup Relay Test Interval Extension".

B 3.3 INSTRUMENTATION

B 3.3.7 Diesel Generator (DG)-Loss of Voltage Start (LOVS)

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOVS in the event a Loss of Voltage (LOV) or Degraded Voltage (DV) condition occurs. There is one LOVS for each 4.16 kV vital bus.

Four solid-state degraded voltage relays and four undervoltage relays with inverse time characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage, respectively. The Loss Of Voltage relays generate a LOVS if the voltage is below 70% for a short time or below 78% for a longer time. The Degraded Voltage relays generate a LOVs if voltage is below 90% for a long time. The Balance of Plant Engineered Safety Features Activation System (BOP ESFAS) Loss of Power/Load Shed (LOP/LS) module receives inputs from the LOV and DV relays. The LOP/LS module has four channels, each of the channels has one LOV input and one DV input. If either a LOV or DV signal is received in that channel, the channel trips. If any 2 of the 4 channels trip, a signal is sent to the BOP ESFAS Diesel Generator Start Signal (DGSS) module starting the diesel. The LOVS initiated actions are described in "Onsite Power Systems" (Ref. 1).

Trip Setpoints and Allowable Values

The trip setpoints and Allowable Values are based on the analytical limits presented in "Onsite Power Systems," Reference 1. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in SR 3.3.7.3 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 3. The actual nominal trip setpoint is normally still more

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

conservative than that required by the plant specific setpoint calculations. If the measured trip setpoint does not exceed the documented Surveillance acceptance criteria, the undervoltage relay is considered OPERABLE.

Setpoints in accordance with the Allowable Values will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

The undervoltage protection scheme has been designed to protect the plant from spurious trips caused by the offsite power source. A complete loss of offsite power will result in approximately a 2 second delay in LOVS actuation. The DG starts and is available to accept loads within a 10 second time interval on the Engineered Safety Features Actuation System (ESFAS) or LOVS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

Since there are four protective channels in a two-out-of-four trip logic for each division of the 4.16 kV power supply, no single sensor failure will cause or prevent protective system actuation.

APPLICABLE SAFETY ANALYSES

The DG-LOVS is required for Engineered Safety Features (ESF) systems to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Accident analyses credit the loading of the DG based on a loss of offsite power during a loss of coolant accident. The actual DG start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analysis assumes a nonmechanistic DG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOVS, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed. LOVS channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 4).

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

The DG-LOVS channels satisfy Criterion 3 of 10 CFR 50.36(C)(2)(ii).

LCO

The LCO for the LOVS requires that four channels per bus of LOVS instrumentation be OPERABLE in MODES 1, 2, 3, and 4 and when the associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources-Shutdown." The LOVS supports safety systems associated with the ESFAS. In MODES 5 and 6, the four channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed.

(continued)

50-528 PALO VERDE 1 APSCO

APS COMMENTS ON THE DRAFT NRC SAFETY EVALUATION FOR PROPOSED CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS BOOK #3

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BASES

LCO (continued)

Actions allow maintenance (trip channel) bypass of individual channels.

Loss of LOVS Function could result in the delay of safety system initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is an anticipated operational occurrence, the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only the one turbine-driven pump as well as an increased potential for a loss of decay heat removal through the secondary system.

Only Allowable Values are specified for each Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable, provided that operation and testing is consistent with the assumptions of the plant specific setpoint calculation. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

APPLICABILITY

The DG-LOVS actuation Function is required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE, so that it can perform its function on a loss of power or degraded power to the vital bus.

BASES

ACTIONS

A LOVS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the instrument is set up for adjustment to bring it within specification. If the actual trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

A.1 and A.2

Condition A applies if one channel per DG bus is inoperable.

If the channel cannot be restored to OPERABLE status, the affected channel should either be bypassed or tripped within 1 hour (Required Action A.1).

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Placing this channel in either Condition ensures that logic is in a known configuration. In trip, the LOVS Logic is one-out-of-three. In bypass, the LOVS Logic is two-out-of-three. The 1 hour Completion Time is sufficient to perform these Required Actions.

Once Required Action A.1 has been complied with, Required Action A.2 allows prior to entering MODE 2 following the next MODE 5 entry to repair the inoperable channel. If the channel cannot be restored to OPERABLE status, the plant cannot enter MODE 2 following the next MODE 5 entry. The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The prior to entering MODE 2 following the next MODE 5 entry Completion Time is based on adequate channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a system actuation.

B.1 and B.2

Condition B applies if two channels per DG bus are inoperable.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protection system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protection system operation.

If the channel cannot be placed in bypass or trip within 1 hour, the Conditions and Required Actions for the associated DG made inoperable by DG – LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2. This places the Function in one-out-of-two logic. The 1 hour Completion Time is sufficient to perform the Required Actions.

(continued)

ACTIONS

B.1 and B.2 (continued)

One of the two inoperable channels will need to be restored to OPERABLE status prior to the next required CHANNEL FUNCTIONAL TEST because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not permitted to bypass more than one DG-LOVS channel, and placing a second channel in trip will result in a loss of voltage diesel start signal.

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel.

C.1

Condition C.1 applies when more than two channels on a single bus are inoperable.

Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour. With more than two channels inoperable, the logic is not capable of providing the DG-LOVS signal for valid Loss of Voltage or degraded voltage condition. The 1 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.

D.1

Condition D.1 applies if the Required Actions and associated Completion Times are not met.

Required Action D.1 ensures that Required Actions for the affected DG inoperabilities are initiated. Depending upon plant MODE, the ACTIONS specified in LCO 3.8.1, "AC Sources-Operating," or LCO 3.8.2 are required immediately.

(continued)

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG-LOVS Function.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a 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. A CHANNEL CHECK consists of verifying all relay status lights on the control board are lit. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based upon operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.2

A CHANNEL FUNCTIONAL TEST is performed every 18 months to ensure that the entire channel will perform its intended function when needed.

The Frequency of 18 months is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 18 months Frequency is a rare event. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.2 (continued)

The as found and as left values must also be recorded and reviewed for consistency.

SR 3.3.7.3

SR 3.3.7.3 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel. This includes calibration of the Loss of Voltage and Degraded Voltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive surveillances to ensure the instrument channel remains operational. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency.

The setpoints, as well as the response to a Loss of Voltage and Degraded Voltage test, shall include a single point verification that the trip occurs within the required delay time, as shown in Reference 1. The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 8.3
2. UFSAR, Chapter 15.
3. Controlled Dwg. Relay Setpoint Sheets.
4. 10 CFR 50, Appendix A, GDC 21.

B 3.3 INSTRUMENTATION

B 3.3.8 Containment Purge Isolation Actuation Signal (CPIAS)

BASES

BACKGROUND

This LCO encompasses the CPIAS, which is an instrumentation channel that performs an actuation function required for plant protection but is not otherwise included in LCO 3.3.6, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.7, "Diesel Generator (DG)–Loss of Voltage Start (LOVS)."

The CPIAS provides protection from radioactive contamination in the containment in the event a fuel assembly should be severely damaged during handling. It also closes the purge valves during plant operation in response to a Reactor Coolant System (RCS) leak.

The CPIAS will detect any abnormal amounts of radioactive material in the power access and refueling purge exhaust ducts and will initiate purge valve closure to limit the release of radioactivity to the environment. Both the power access purge and refueling purge supply and exhaust valves are closed on a CPIAS when a high radiation level in the power access and refueling purge exhaust ducts is detected.

The CPIAS includes two independent, redundant logic subsystems, including actuation trains. Each train employs a Gamma (area) sensor.

If either sensor exceeds the trip setpoint, both of the CPIAS trains will be actuated (one-out-of-two logic).

Each train actuates a separate series valve in the containment purge supply and return lines. Either train controls sufficient equipment to perform the isolation function. These valves are also isolated on a Containment Isolation Actuation Signal (CIAS).

(continued)

BASES

BACKGROUND (continued)

Trip Setpoints and Allowable Values

Trip setpoints used in the bistables are based on the analytical limits (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The trip setpoints are digitally generated by the radiation monitors. These trips values are not subject to drifts common to trips generated by analog type equipment. The allowable value for this trip is therefore the same as the Trip Setpoints.

Setpoints in accordance with the Allowable Value will ensure that the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CPIAS is a backup to the CIAS Systems in MODES 1, 2, 3, and 4 and will close the containment purge valves in the event of high radiation levels resulting from a primary leak in the containment.

The CPIAS is also required to close the containment purge valves in the event of the fuel handling accident in containment, as described in Reference 1. This accident is a limiting case representing a class of accidents that might involve radiation release in containment without CIAS actuation. This minimizes the offsite consequences of radiation accidents in containment. The CPIAS however, is not required to function to ensure the offsite consequences of radiation accidents in containment are within 10 CFR 100 limits (Ref. 2) as described in the Safety Analysis (Ref. 1).

The CPIAS satisfies the requirements of Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

BASES

LCO

LCO 3.3.8 requires one CPIAS channel to be OPERABLE. The required channel consists of an area radiation monitor; Actuation Logic; and Manual Trip. The specific trip setpoints for the CPIAS are listed in the SRs.

Each trip setpoint specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the trip function.

The Bases for the LCO on CPIAS are discussed below for each Function:

a. Manual Trip

The LCO on Manual Trip backs up the automatic trip and ensures operators have the capability to rapidly initiate the CPIAS Function if any parameter is trending toward its setpoint. One manual channel of CPIAS is required in MODES 1, 2, 3, and 4, since the CPIAS is redundant with the CIAS and there are additional means of closing the containment purge valves. Only one manual channel of CPIAS is required during CORE ALTERATIONS and movement of irradiated fuel assemblies, since there are additional means of closing the containment purge valves in the event of a channel failure.

b. Power Access and Refueling Purge Exhaust Duct Radiation

One channel of radiation monitoring is required during in MODES 1, 2, 3, and 4 or during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment.

(continued)

BASES

LCO (continued)

c. Actuation Logic

One channel of Actuation Logic is required since the valves can be shut independently of the CPIAS signal either manually from the control room or using the CIAS pushbutton.

APPLICABILITY

In MODES 1, 2, 3, and 4, the power access purge valves may be open. In these MODES, it is necessary to ensure the valves will shut in the event of a primary leak in containment whenever any of the containment purge valves are open.

With the purge valves open during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, there is the possibility of a fuel handling accident requiring CPIAS on high radiation in the power access purge and refueling purge exhaust ducts.

The Applicability is modified by a Note, which states that the CPIAS specification is only required when the penetration is not isolated by at least one closed automatic valve, closed manual valve, or blind flange.

ACTIONS

A CPIAS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure.

A.1

Condition A applies to the failure of CPIAS Manual Trip, Actuation Logic, and area radiation monitor in MODES 1, 2, 3, and 4. The Required Action is to place and maintain containment purge and exhaust valves in closed position. The Completion Time accounts for the condition that the capability to isolate containment on valid high radiation levels in the power access and refueling purge exhaust ducts or manual signals is degraded during power operation or shutdown modes.

(continued)

BASES

ACTIONS (continued)

B.1

Condition B applies when the Required Action and associated Completion Time of Condition A are not met in MODES 1, 2, 3, or 4. If Required Action A cannot be met within the required Completion Time, entry into LCO 3.6.3 "Containment Isolation Valves" is required. The Completion Time accounts for the fact that the inability to close and maintain the purge and exhaust valves closed may affect the ability of the valves to automatically close on a Containment Isolation Actuation Signal (CIAS)

C.1, C.2.1, and C.2.2.

Condition C applies to two channels of radiation monitor, Manual Trip, or Actuation Logic inoperable, the applicability is during CORE ALTERATIONS or during the movement of irradiated fuel assemblies within containment. Required Action C.1 is to place the containment purge and exhaust isolation valves in the closed position. The Required Action immediately performs the isolation function of the CPIAS. Required Actions C.2.1 and C.2.2 may be performed in lieu of Required Action C.1. Required Action C.2.1 requires the suspension of CORE ALTERATIONS and Required Action C.2.2 requires suspension of movement of irradiated fuel in containment immediately. The Completion Time accounts for the fact that the automatic capability to isolate containment on valid power access and refueling purge exhaust duct high radiation signals is degraded during conditions in which a fuel handling accident is possible and CPIAS provides the only automatic mitigation of radiation release.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred on the required radiation monitor channels used in the CPIAS. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A CHANNEL FUNCTIONAL TEST is performed on the required containment radiation monitoring channel to ensure the entire channel will perform its intended function. The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92 day Frequency is a rare event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

Proper operation of the individual actuation relays is verified by actuating these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic every 18 months. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified. The Frequency of 18 months is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function during any 18 month interval is a rare event. A Note to the SR indicates that this Surveillance includes verification of operation for each actuation relay.

SR 3.3.8.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The 18 month Frequency is based on plant operating experience with regard to channel OPERABILITY and drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.5

Every 18 months, a CHANNEL FUNCTIONAL TEST is performed on the CPIAS Manual Trip channel.

This test verifies that the trip handswitches are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing manual actuation of the Function. The 18 month Frequency is based on operating experience that has shown these components usually pass the Surveillance when performed at a Frequency of once every 18 months.

REFERENCES

1. UFSAR, Chapter 15.
2. 10 CFR 100.

B 3.3 INSTRUMENTATION

B 3.3.9 Control Room Essential Filtration Actuation Signal (CREFAS)

BASES

BACKGROUND

This LCO encompasses CREFAS actuation, which is an instrumentation channel that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.6, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.7, "Diesel Generator (DG)—Loss of Voltage Start (LOVS)." This is a BOP ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCO 3.3.6 and LCO 3.3.7.

The CREFAS initiates actuation of the Control Room Essential Filtration System to minimize operator radiation exposure. The CREFAS includes two independent, redundant subsystems, including actuation trains. Each train has a gaseous activity radiation monitor for the control room air intake activity. If either train radiation monitor indicates an unsafe condition, both CREFAS trains will be actuated (one-out-of-two logic). The two trains actuate separate equipment. Actuating either train will perform the intended function. A CREFAS is also initiated by a Containment Purge Isolation Actuation Signal (CPIAS) from either of the two CPIAS channels or by a Fuel Building Essential Ventilation Actuation Signal (FBEVAS) from either of the two FBEVAS channels. Control room filtration also occurs on a Safety Injection Actuation Signal (SIAS).

Trip Setpoints and Allowable Values

Trip setpoints used in the bistables are based on the analytical limits (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The trip setpoints are digitally generated by the radiation monitors. These trip values are not subject to drifts common to analog type equipment. The allowable value for this trip is therefore the same as the trip setpoint.

(continued)

BASES (continued)

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Setpoints in accordance with the Allowable Value will ensure that the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CREFAS maintains the control room atmosphere within conditions suitable for prolonged occupancy throughout the duration of any one of the accidents discussed in Reference 1. The radiation exposure of control room personnel, through the duration of any one of the postulated accidents discussed in "Accident Analysis," FSAR, Chapter 15 (Ref. 1), does not exceed the limits set by 10 CFR 50, Appendix A, GDC 19 (Ref. 2).

The CREFAS satisfies the requirements of Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

LCO 3.3.9 requires one channel of CREFAS to be OPERABLE. The required channel consists of Actuation Logic, Manual Trip, and a gaseous radiation monitor. The specific trip setpoint for the CREFAS is listed in the SR.

Each trip setpoint specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the trip Function. A channel is inoperable if its actual trip setpoint is not set to the value specified in SR 3.3.9.2.

The Bases for the LCO on the CREFAS are discussed below for each Function:

(continued)

BASES

LCO (continued)

a. Manual Trip

The LCO on Manual Trip backs up the automatic trips and ensures operators have the capability to rapidly initiate the CREFAS Function if any parameter is trending toward its setpoint. One channel must be OPERABLE. This considers that the Manual Trip capability is a backup and that other means are available to actuate the redundant train if required, including manual SIAS, FBEVAS, or CPIAS.

b. Radiation Monitors

One channel of radiation monitor is required to be OPERABLE to ensure the control room filtration actuates on high gaseous activity.

c. Actuation Logic

One train of Actuation Logic must be OPERABLE, since there are alternate means available to actuate the redundant train, including SIAS.

APPLICABILITY

The CREFAS Functions must be OPERABLE in MODES 1, 2, 3, 4, 5, and 6 and during movement of irradiated fuel assemblies in either the fuel building or the containment building, to ensure a habitable environment for the control room operators.

The CREFAS must be OPERABLE in MODES 5 and 6 to provide protection for a Waste Gas Decay Tank rupture accident.

ACTIONS

A CREFAS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure.

(continued)

BASES

ACTIONS (continued)

A.1, B.1, B.2, C.1, C.2.1, C.2.2, and C.2.3

Conditions A, B, and C are applicable to manual and automatic actuation of the CREFAS. Condition A applies to the failure of two channels of the CREFAS Manual Trip, Actuation Logic, and radiation monitor channel in MODE 1, 2, 3, or 4. Entry into this Condition requires action to either restore the failed channel or manually perform the CREFS safety function (Required Action A.1). The Completion Time of 1 hour is sufficient to complete the Required Actions and accounts for the fact that CREFAS supplements control room filtration by other Functions (e.g., SIAS) in MODES 1, 2, 3, and 4. If Required Action A.1 and the associated completion time are not met, 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 (Required Action B.1) and to MODE 5 within 36 hours (Required Action B.2). The Completion Times of 6 hours and 36 hours for reaching MODES 3 and 5 from MODE 1 are reasonable, based on operating experience and normal cooldown rates, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant safety systems or operators.

Condition C applies to the failure of two channels of CREFAS Manual Trip, Actuation Logic, and radiation monitor channel in MODE 5 or 6, or when moving irradiated fuel assemblies. The Required Actions are immediately taken to place one OPERABLE CREFS train in operation or to suspend CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel assemblies. The Completion Time recognizes the fact that FBEVAS, or CPIAS are available to initiate control room filtration in the event of a fuel handling accident.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.9.2

A CHANNEL FUNCTIONAL TEST is performed on the required control room radiation monitoring channel to ensure the entire channel will perform its intended function.

The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92 day interval is a rare event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.3

Proper operation of the individual actuation relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic every 18 months. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified.

The Frequency of 18 months is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any 18 month interval is a rare event.

Note 1 indicates this Surveillance includes verification of operation for each actuation relay.

Note 2 indicates that relays that cannot be tested at power are excepted from the Surveillance Requirement while at power. These relays must, however, be tested during each entry into MODE 5 exceeding 24 hours unless they have been tested within the previous 6 months. At PVNGS all of the actuation relays can be tested at power.

SR 3.3.9.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.5

Every 18 months, a CHANNEL FUNCTIONAL TEST is performed on the manual CREFAS actuation circuitry. This test verifies that the trip handswitches are capable of opening contacts in the Actuation Logic as designed, de-energizing the actuation relays and providing Manual Trip of the function. The 18 month Frequency is based on the operating experience that has shown these components usually pass the Surveillance when performed at a Frequency of once every 18 months.

SR 3.3.9.6

This Surveillance ensures that the train actuation response times are less than the maximum times assumed in the analyses. Response time testing criteria are included in Reference 3. ESF Response Time tests are conducted on a Staggered Test Basis of once per 18 months. The 18 month Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Testing of the final actuating devices, which make up the bulk of the response time, is included in the Surveillance testing.

REFERENCES

1. UFSAR, Chapter 15.
2. 10 CFR 50, Appendix A, GDC 19.
3. UFSAR, Chapter 7.

B 3.3 INSTRUMENTATION

B 3.3.10 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by plant specific documents (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3).

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and

(continued)

BASES

BACKGROUND (continued)

- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

These key variables are identified by plant specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identified the plant specific Type A variables and provided justification for deviating from the NRC proposed list of Category I variables.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs; and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I, non-Type A variables are important in reducing public risk.

LCO

LCO 3.3.10 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

The exception to the two channel requirement is Containment Isolation Valve Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.10-1.

1. Logarithmic Neutron Flux

   Logarithmic Neutron Flux indication is provided to verify reactor shutdown.

(continued)

BASES

LCO

1. Logarithmic Neutron Flux (continued)

At PVNGS, the Logarithmic Neutron Flux PAM channels consist of the following:

SEA-NE-001A
SEB-NE-001B

2, 3. Reactor Coolant System (RCS) Hot and Cold Leg Temperature

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

Reactor outlet temperature inputs to the PAM are provided by two fast response resistance elements and associated transmitters in each loop. Cold Legs 1A and 1B make up one loop and Cold Legs 2A and 2B make up one loop. The channels provide indication over a range of 50°F to 750°F.

At PVNGS the Hot Leg Temperature indication consists of:

RCA-TT-112H1
RCB-TT-112H2
RCA-TT-122H1
RCB-TT-122H2

The Cold Leg Temperature indication consists of:

RCA-TT-112C1
RCB-TT-112C2
RCA-TT-122C1
RCB-TT-122C2

4. Reactor Coolant System Pressure (wide range)

RCS Pressure (wide range) is a Category I variable, provided for verification of core cooling and RCS integrity long term surveillance.

(continued)

BASES

LCO

4. Reactor Coolant System Pressure (wide range) (continued)

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 4000 psig. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the Qualified Safety Parameter Display System (QSPDS) plasma display. The QSPDS plasma display is the primary indication used by the operator during an accident. Therefore, the PAM instrumentation Specification deals specifically with this portion of the instrument channel.

RCS pressure is also a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture or small break loss of coolant accident (LOCA). Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate reactor coolant pump operation.

At PVNGS the RCS Pressure (wide range) consists of:

RCA-PT-190A
RCB-PT-190B

5. Reactor Vessel Water Level

Reactor Vessel Water Level is provided for verification and long term surveillance of core cooling.

The Reactor Vessel Water Level Monitoring System provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

(continued)

BASES

LCO

5. Reactor Vessel Water Level (continued)

The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval. The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

At PVNGS the Reactor Vessel Water Level is displayed on QSPDS A and QSPDS B.

6. Containment Sump Water Level (wide range)

Containment Sump Water Level is provided for verification and long term surveillance of RCS integrity.

At PVNGS, Containment Sump Water Level instrumentation consists of the following:

SIA-LT-706
SIB-LT-707

7. Containment Pressure (wide range)

Containment Pressure is provided for verification of RCS and containment OPERABILITY.

At PVNGS, Containment Pressure instrumentation consists of the following:

HCA-PT-353A
HCB-PT-353B

(continued)

BASES

LCO (continued)

8. Containment Isolation Valve Position

Containment Isolation Valve Position is provided for verification of containment OPERABILITY.

CIV position is provided for verification of containment integrity. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

The PVNGS design uses three indications for each valve that receives an automatic close signal from the ESFAS. Each of these indications use a different contact on the position switch. One contact provides an open/close indication on the valve control handswitch in the main control room. This indication uses the same Class 1E power that is used by the valve control circuit. A second contact is used by the Safety Equipment Status System (SESS). This system receives inputs from each valve and the ESFAS system. After an ESFAS actuation any valve that does not reposition to the fully closed position is indicated and annunciated in the main control room. There are two channels of SESS, one channel receives power from the A Train Class 1E DC Bus and indicates the status of the A Train actuated equipment, and one channel receives power from the B Train Class 1E DC Bus and indicates the status of the B Train actuated equipment.

(continued)

BASES

LCO

8. Containment Isolation Valve Position (continued)

The third contact provides an indication of valve position to the Emergency Response Facility Data Acquisition and Display System (ERFDADS). This signal is Class 1E until it goes through a qualified isolator.

The ERFDADS computer and displays are non-Class 1E. For the purpose of this Specification either the SESS indication or the handswitch indication in the main control room may be used.

At PVNGS the Containment Isolation Valve position instrumentation consists of:

CPA-UV-2A	Containment Refueling Purge Supply
CPA-UV-2B	Containment Refueling Purge Exhaust
CPB-UV-3A	Containment Refueling Purge Supply
CPB-UV-3B	Containment Refueling Purge Exhaust
CPA-UV-4A	Containment Power Access Purge Supply
CPA-UV-4B	Containment Power Access Purge Exhaust
CPB-UV-5A	Containment Power Access Purge Supply
CPB-UV-5B	Containment Power Access Purge Exhaust
CHB-UV-505	RCP Controlled Bleedoff to VCT
CHA-UV-506	RCP Controlled Bleedoff to VCT
CHA-UV-516	Letdown to Regen HX
CHB-UV-523	Letdown from Regen HX
CHA-UV-560	Reactor Drain Tank Outlet
CHB-UV-561	Reactor Drain Tank Outlet
CHA-UV-580	Make-Up Supply to Reactor Drain Tank
CHA-UV-715	Sample Return to Reactor Drain Tank
CHB-UV-924	Letdown Line Sample PASS
GAA-UV-1	HP Nitrogen to Safety Injection Tanks
GAA-UV-2	LP Nitrogen to Containment
GRA-UV-1	Waste Gas Header
GRB-UV-2	Waste Gas Header

(continued)

BASES

LCO

8. Containment Isolation Valve Position (continued)

HCB-UV-44	Radiation Monitor RU-1 Supply
HCA-UV-45	Radiation Monitor RU-1 Supply
HCA-UV-46	Radiation Monitor RU-1 Return
HCB-UV-47	Radiation Monitor RU-1 Return
HPA-UV-1	Containment Hydrogen Control System
HPB-UV-2	Containment Hydrogen Control System
HPA-UV-3	Hydrogen Recombiner Supply
HPB-UV-4	Hydrogen Recombiner Supply
HPA-UV-5	Hydrogen Recombiner Return
HPB-UV-6	Hydrogen Recombiner Return
HPA-UV-23	Hydrogen Monitor Return
HPA-UV-24	Hydrogen Monitor Supply
IAA-UV-2	Instrument and Service Air
NCB-UV-401	Nuclear Cooling Water
NCA-UV-402	Nuclear Cooling Water
NCB-UV-403	Nuclear Cooling Water
RDA-UV-23	Containment Sumps
RDB-UV-24	Containment Sumps
RDB-UV-407	Containment Radwaste Sumps
SGB-HV-200	Steam Generator #1 Chemical Injection
SGB-HV-201	Steam Generator #2 Chemical Injection
SIA-UV-708	Containment Recirc Sump PASS
SSB-UV-200	Hot Leg Sample PASS
SSB-UV-201	Surge Line PASS
SSB-UV-202	Pressurizer Steam Space PASS
SSA-UV-203	Hot Leg Sample PASS
SSA-UV-204	Surge Line PASS
SSA-UV-205	Pressurizer Steam Space PASS
WCB-UV-61	Normal Chilled Water Return Header
WCA-UV-62	Normal Chilled Water Return Header
WCB-UV-63	Normal Chilled Water Supply Header

(continued)

BASES

LCO (continued)

9. Containment Area Radiation (high range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

At PVNGS, Containment Area Radiation instrumentation consists of the following:

SQA-RU-148
SQB-RU-149

10. Containment Hydrogen Monitors

Containment Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions.

At PVNGS, Containment Hydrogen instrumentation consists of the following:

HPA-AI-9
HPB-AI-10

11. Pressurizer Level

Pressurizer Level is used to determine whether to terminate Safety Injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition.

At PVNGS, Pressurizer Level instrumentation consists of the following:

RCA-LT-110X
RCB-LT-110Y

(continued)

BASES

LCO (continued)

12. Steam Generator Water Level

Steam Generator Water Level is provided to monitor operation of decay heat removal via the steam generators. The Category I indication of steam generator level is the wide range level instrumentation. The wide range level covers a span of 143 inches above the lower tubesheet to 55.5 inches above the steam separator deck.

Wide Range Steam Generator Level is a Type A variable because the operator must manually control steam generator level during a Steam Generator Tube Rupture (STGR) event to ensure steam generator tube coverage. At PVNGS wide range Steam Generator Level Instrumentation consists of:

SGA-LT-1113A
SGB-LT-1113B
SGC-LT-1113C
SGD-LT-1113D

SGA-LT-1123A
SGB-LT-1123B
SGC-LT-1123C
SGD-LT-1123D

13. Condensate Storage Tank (CST) Level

CST Level is provided to ensure water supply for AFW. The CST provides the ensured, safety grade water supply for the AFW System. Inventory is monitored by a 0 to 100% level indication. CST Level is displayed on a control room indicator.

At PVNGS CST Level Instrumentation consists of:

CTA-LT-35
CTB-LT-36

(continued)

BASES

LCO
(continued)

14, 15, 16, 17. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for inadequate core cooling detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures.

Based on these evaluations, adequate or inadequate core cooling detection is ensured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 61 incore instrument detector assemblies.

The junction of each thermocouple is located a few inches above the fuel assembly, inside a structure that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

(continued)

BASES

LCO (continued)

18. Steam Generator Pressure

Steam Generator pressure indication is provided for Steam Generator pressure verification. At PVNGS Steam Generator Pressure Instrumentation consists of:

SGA-PT-1013A
SGB-PT-1013B
SGC-PT-1013C
SGD-PT-1013D

SGA-PT-1023A
SGB-PT-1023B
SGC-PT-1023C
SGD-PT-1023D

19. Reactor Coolant System-Subcooling Margin Monitoring

The RCS Subcooling Margin Monitor is a portion of the Inadequate Core Cooling (ICC) Instrumentation required by Item II.F.2 in NUREG-0737, the post-TMI Action Plan. The ICC instrumentation enhances the ability of the Operator to anticipate the approach to, and recovery from, ICC. At PVNGS RCS subcooling Margin Monitoring Instrumentation consists of:

QSPDS A
QSPDS B

20. Reactor Coolant System Activity

The RCS Activity provides an indication of fuel cladding failure. This indicates degradation of the first of three barriers to fission product release to the environment. The three barriers to fission product release are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment. At PVNGS the RCS Activity Instrumentation consists of:

SQA-RU-150
SQB-RU-151

(continued)

BASES

LCO (continued)

Two channels are required to be OPERABLE for all but one Function. Two OPERABLE channels ensure that no single failure within the PAM instrumentation or its auxiliary supporting features or power sources, concurrent with failures that are a condition of or result from a specific accident, prevents the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident.

In Table 3.3.10-1 the exception to the two channel requirement is Containment Isolation Valve Position.

Two OPERABLE channels of core exit thermocouples are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Plant specific evaluations in response to Item II.F.2 of NUREG-0737 (Ref. 3) have determined that any two thermocouple pairings per quadrant, satisfy these requirements. Two sets of two thermocouples in each quadrant ensure a single failure will not disable the ability to determine the radial temperature gradient.

For loop and steam generator related variables, the required information is individual loop temperature and individual steam generator level. In these cases two channels are required to be OPERABLE for each loop of steam generator to redundantly provide the necessary information.

(continued)

BASES

LCO (continued)

In the case of Containment Isolation Valve Position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to monitor an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.10-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS (continued)

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.6, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation.

(continued)

BASES

ACTIONS

C.1 (continued)

Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This Required Action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.10-1. The applicable Condition referenced in the Table is Function dependent. Each time Required Action C.1 or D.1 is not met, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Condition C are not met and Table 3.3.10-1 directs entry into Condition F, 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.

(continued)

BASES

ACTIONS (continued)

G.1

Alternate means of monitoring Reactor Vessel Water Level, RCS Activity, and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.6. The report provided to the NRC should discuss whether the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the SR table specifies that the following SRs apply to each PAM instrumentation Function found in Table 3.3.10-1.

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.10.2

A CHANNEL CALIBRATION is performed every 18 months or approximately every refueling. CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note excludes the neutron detectors from the CHANNEL CALIBRATION.

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION as described in UFSAR Sections 18.II.F.1.3 and 11.5.2.1.6.2 will be performed.

The calibration of the Containment Isolation Valve (CIV) position indication channels will consist of verification that the position indication changes from not-closed to closed when the valve is actuated to its isolation position by SR 3.6.3.7. The position switch is the sensor for the CIV position indication channels.

The calibration of the containment hydrogen monitor will use sample gases containing a nominal one volume percent hydrogen, balance nitrogen, and four volume percent hydrogen, balance nitrogen.

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift.

BASES

REFERENCES

1. UFSAR Section 1.8, Table 1.8-1.
2. Regulatory Guide 1.97, Revision 2.
3. NUREG-0737, Supplement 1.

B 3.3 INSTRUMENTATION

B 3.3.11 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator safety valves or the steam generator atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allow extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to bring the plant to, and maintain it in, MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down the plant and maintain it in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 50, Appendix A, GDC 19 (Ref. 1) and Appendix R (Ref. 2).

The Remote Shutdown System has been identified as an important contributor to the reduction of plant accident risk and, therefore, has been retained in the Technical Specifications, as indicated in 10 CFR 50.36 (c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation required is listed in Table 3.3.11-1 in the accompanying LCO. The disconnect switches and control circuits are listed in PVNGS controlled documents.

The controls, instrumentation, and transfer switches are those required for:

- Reactivity Control (initial and long term);
- RCS Pressure Control;
- Decay Heat Removal;
- RCS Inventory Control; and
- Safety support systems for the above Functions, as well as the essential spray pond system, essential cooling water system, and onsite power including the diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the remote shutdown Functions are OPERABLE.

The Remote Shutdown System instrumentation and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instrument and control circuits will be OPERABLE if plant conditions require that the Remote Shutdown System be placed in operation.

(continued)

BASES

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control become unavailable.

ACTIONS

A Note has been included that excludes the MODE change restrictions of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require a plant shutdown. This is acceptable due to the low probability of an event requiring this system.

A Remote Shutdown System division is inoperable when each Function listed in Table 3.3.11-1 is not accomplished by the required number of channels in Table 3.3.11-1 that satisfies the OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.11-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the situation where one or more instrumentation channels of the Remote Shutdown System are inoperable. This includes any Function listed in Table 3.3.11-1.

(continued)

BASES

ACTIONS

A.1 (continued)

The Required Action is to restore the channels to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

Condition B addresses the situation where one or more disconnect or control circuits of the Remote Shutdown System are inoperable. The required disconnect and control circuits are listed in PVNGS controlled documents.

The required Action is to restore the required switch(s)/circuit(s) to OPERABLE status or issue procedure changes that identify alternate disconnect methods or control circuits. The Completion Time for either of the two Actions is 30 days.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A are not met, 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 MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized.

If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are offscale in the same direction. Current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.11.2

SR 3.3.11.2 verifies that each required Remote Shutdown System transfer switch and control circuit performs its intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be brought to and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels seldom fail to pass the Surveillance when performed at a Frequency of once every 18 months.

SR 3.3.11.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to the measured parameter within the necessary range and accuracy.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. 10 CFR 50, Appendix R.

B 3.3 INSTRUMENTATION

B 3.3.12 Boron Dilution Alarm System (BDAS)

BASES

BACKGROUND

The Boron Dilution Alarm System (BDAS) alerts the operator of a boron dilution event in MODES 3, 4 and 5. The boron dilution alarm is received at least 15 minutes prior to criticality to allow the operator to terminate the boron dilution.

In MODES 1 and 2 protection for a boron dilution event is addressed by LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation-Operating." In MODES 3 and 4 with the CEAs withdrawn, LCO 3.3.2, " Reactor Protective System (RPS) Instrumentation-Shutdown," provides protection. In MODE 6 protection for a boron dilution event is addressed by LCO 3.9.2, "Refueling Operations - Nuclear Instrumentation."

The BDAS utilizes two channels that monitor the startup channel neutron flux indications. If the neutron flux signals increase to the calculated alarm setpoint a control room annunciation is received. The setpoint is automatically lowered to a fixed amount above the current flux level signal. The alarm setpoint will only follow decreasing or constant flux levels, not increasing levels. Two channels of BDAS must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.

APPLICABLE SAFETY ANALYSES

The BDAS channels are necessary to monitor core reactivity changes. They are the primary means for detecting and triggering operator actions to respond to boron dilution events initiated from conditions in which the RPS is not required to be OPERABLE.

The OPERABILITY of BDAS channels is necessary to meet the assumptions of the safety analyses to mitigate the consequences of an inadvertent boron dilution event as described in the UFSAR, Chapter 15 (Ref. 1).

The BDAS channels satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

The LCO on the BDAS channels ensures that adequate information is available to mitigate the consequences of a boron dilution event.

A minimum of two BDAS channels are required to be OPERABLE. Because the BDAS utilizes the excore startup channel instrumentation as its detection system the OPERABILITY of the excore startup channel is also part of the OPERABILITY of the BDAS.

APPLICABILITY

The BDAS must be OPERABLE in MODES 3, 4, and 5 because the safety analysis assumes this alarm will be available in these MODES to alert the operator to take action to terminate the boron dilution. In MODES 1 and 2, and in MODES 3, 4, and 5, with the RTCBs shut and the CEAs capable of withdrawal, the logarithmic power monitoring channels are addressed as part of the RPS in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation—Operating" and LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation-Shutdown".

The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation." The excore startup channels provide neutron flux coverage extending an additional one to two decades below the logarithmic channels for use during shutdown and refueling, when neutron flux may be extremely low.

The Applicability is modified by a Note that the BDAS is required in MODE 3 within 1 hour after the neutron flux is within the startup range following a reactor shutdown. This allows the neutron flux level to decay to a level within the range of the excore startup channels and for the operator to initialize the BDAS.

(continued)

BASES (continued)

ACTIONS

A channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section of the Bases.

A.1

With one required channel inoperable, Required Action A.1 requires the RCS boron concentration to be determined immediately and at the applicable monitoring Frequency specified in the COLR. The RCS boron concentration may be determined by the boronometer reading or by RCS sampling. The RCS sample should be from the hot leg if one or more Reactor Coolant Pumps (RCPs) are running or from the discharge of the operating pump providing shutdown cooling flow with no RCPs running. The monitoring Frequency specified in the COLR ensures that a decrease in the boron concentration during a boron dilution event will be detected. The boron concentration measurement and the OPERABLE BDAS channel provide alternate methods of detection of boron dilution with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

With two required channels inoperable Required Action B.1 requires the RCS boron concentration to be determined by redundant methods immediately and at the monitoring Frequency specified in the COLR. The redundant methods may use the boronometer and RCS sampling or independent collection and analysis of two RCS samples. The RCS sample should be from the hot leg if one or more Reactor Coolant Pumps (RCPs) are running or from the discharge of the operating pump providing shutdown cooling flow with no RCPs running. The simultaneous use of the boronometer and RCS sampling or independent collection and analysis of two RCS samples to monitor the RCS boron concentration provides alternate indications of inadvertent boron dilution. This will allow detection with sufficient time for termination of boron dilution before complete loss of SHUTDOWN MARGIN and return to criticality.

C.1

Condition C is entered when the Required Actions and associated Completion Times of Condition A or B are not met. If the Required Actions associated with these Conditions cannot be completed within the required Completion Time, the neutron flux level monitoring function cannot be reliably performed. The absence of reliable neutron flux level monitoring makes it difficult to ensure SDM is maintained. Required Action C.1 therefore requires that all positive reactivity additions that are under operation control, such as boron dilution or Reactor Coolant System temperature changes, be halted immediately preserving SDM.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

SR 3.3.12.1 is the performance of a CHANNEL CHECK on each required channel every 12 hours. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff and should be based on a combination of the channel instrument uncertainties. If a channel is outside of the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside of its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels.

This SR is modified by a Note that states the CHANNEL CHECK is not required to be performed until 1 hour after neutron flux is within the startup range.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.12.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure that the BDAS is capable of properly alerting the operator to a boron dilution event. Internal excore startup channel test circuitry is used to feed preadjusted test signals into the excore startup channel to verify the proper neutron flux indication is received at the BDAS.

The Frequency is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel in any 92 day Frequency is a rare event. This SR is modified by a Note that states the CHANNEL FUNCTIONAL TEST is not required to be performed until 72 hours after neutron flux is within the startup range. The 72 hours is based on allowing a reasonable time to perform the testing following a plant shutdown.

The CHANNEL FUNCTIONAL TEST of the BDAS consists of online tests including verification of the control room alarm.

SR 3.3.12.3

SR 3.3.12.3 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 18 months. The Surveillance is a complete check and readjustment of the excore startup channel from the input through to the BDAS. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational.

This SR is modified by a Note to indicate that it is not necessary to test the detector, because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output.

REFERENCES

1. UFSAR, Chapter 7 and Chapter 15.

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 DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum Departure from Nucleate Boiling Ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO limits for minimum and maximum RCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limit for minimum and maximum RCS cold leg temperatures are in accordance with the area of acceptable operation shown in Figure 3.4.1-1, are consistent with operation at the indicated power level, and are bounded by those used as the initial temperatures in the analyses.

The LCO limit for minimum RCS flow rate is bounded by those used as the initial flow rates in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.3. This is the acceptance limit for the RCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transients analyzed for include loss of coolant flow events and dropped or stuck Control Element Assembly (CEA) events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Regulating CEA Insertion Limits"; LCO 3.1.8, "Part Length CEA Insertion Limits"; LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI). The safety analyses are performed over the following range of initial values: RCS pressure 2105-2320 psia, core inlet temperature 548-572°F, and reactor vessel inlet coolant flow rate > 95%.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.56(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables—RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate—to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The LCO numerical value for minimum flow rate is given for the measurement location but has not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of minimum flow rate.

APPLICABILITY

In MODE 1 for RCS flow rate, MODES 1 and 2 for RCS pressurizer pressure, Mode 1 for RCS cold leg temperature, and MODE 2 with $K_{eff} \geq 1$ for RCS cold leg temperature, the limits must be maintained during steady state operation in order to ensure that 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 so that DNBR is not a concern.

(continued)

BASES

APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase of > 5% RTP per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core Safety Limits." Those limits are less restrictive than the limits of this LCO, but violation of SLs merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the flow rate is not within the LCO limit, then 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.

The 2 hour Completion Time for restoration of RCS flow rate provides sufficient time to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time 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.

(continued)

BASES

ACTIONS

B.1 (continued)

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of Steam Generator (SG) heat removal.

C.1

Pressurizer pressure and cold leg temperature are controllable and measurable parameter(s). If a parameter is not within the LCO limits, action must be taken to restore the parameter.

The 2 hour Completion Time is based on plant operating experience that shows that these parameter(s) can be restored in this time period.

D.1

If Required Action C.1 is not met within the associated Completion Time, place the plant in MODE 3. In MODE 3 the potential for violation of the DNB limits is greatly reduced.

The 6 hour Completion Time is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of SG heat removal.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action C.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 that 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 verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action C.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for cold leg temperature is sufficient to ensure that the RCS coolant 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.

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 Frequency has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.

This SR is modified by a Note that only requires performance of this SR in MODE 1. The Note is necessary to allow measurement of RCS flow rate at normal operating conditions at power with all RCPs running.

REFERENCES

1. UFSAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies;

b. Operation within the bounds of the existing accident analyses; and

c. Operation with the reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (550°F to 611°F). Nominal T_{cold} for making the reactor critical is 565°F. Safety and operating analyses for lower temperature have not been made.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the 545°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of the LCO is to prevent criticality below the minimum normal operating temperature (550°F) and to prevent operation in an unanalyzed condition.

The LCO is only applicable in MODES 1 and 2 with $K_{eff} \geq 1.0$ and provides a reasonable distance to the limit of 545°F. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.

(continued)

BASES (continued)

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1, and MODE 2 when $K_{eff} \geq 1.0$. Monitoring is required at or below a T_{cold} of 550°F. The no load temperature of 565°F is maintained by the Steam Bypass Control System.

ACTIONS

A.1

If T_{cold} is below 545°F, 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 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

T_{cold} is required to be verified ≥ 545°F once within 30 minutes after any RCS loop T_{cold} < 550°F and every 30 minutes thereafter. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. A Note states the Surveillance is required whenever the reactor is critical and temperature is below 550°F. A second Frequecy requires T_{cold} to be verified within 30 minutes of reaching criticality. This will require repeated performance of SR 3.4.2.1 since a reactor startup takes longer than 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO.

REFERENCES

1. UFSAR, Section 15.

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.

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. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

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. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 2.

(continued)

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 includes the Reference 1 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for inservice leak and hydrostatic (ISLH) testing. However, the criticality limit 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. 5), 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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 follows:

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

(continued)

BASES

LCO (continued)

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification 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, except when reactor vessel head is fully detensioned such that the RCS cannot be pressurized, 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 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.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 to 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. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 5), 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.

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:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress; or

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

b. 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. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < 500 psia within 36 hours.

The 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

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" 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 a short period of time in a controlled manner.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Besides restoring operation to 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 before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 5), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

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 limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency 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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

This SR is modified by a Note that requires this SR be performed only during RCS 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. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

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 RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two Reactor Coolant Pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to Departure from Nucleate Boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with both RCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis had been performed for the four pump combination. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 107% RTP. This is the design overpower condition for four pump operation. The 107% value is the accident analysis setpoint of the nuclear overpower (high flux) trip 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.

RCS Loops – MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36 (C)(2)(ii).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in

(continued)

BASES

LCO (continued)

MODE 3 if any SG level is ≤ 44% wide range level as sensed by the RPS. The minimum water level to declare the SG OPERABLE in MODES 1 or 2 is 44% wide range level.

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, 5, and 6.

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 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

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. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than four RCPs operating.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that the required number of RCS loops are in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. UFSAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops – MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the Steam Generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, Reactor Coolant Pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE.

Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

Analyses have shown that the rod withdrawal event from MODE 3 with one RCS loop in operation is bounded by the rod withdrawal initiated from MODE 2.

Failure to provide 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).

(continued)

BASES

LCO

The purpose of this LCO is to require two RCS loops to be available for heat removal, thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (≥ 25% wide range water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration is prohibited because an even concentration distribution throughout the RCS cannot be ensured. The intent is to stop any known or direct positive reactivity additions to the RCS due to dilution. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODE 3 it is sometimes necessary to stop all RCPs (e.g., to perform surveillance or startup testing, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop (loop 1 or loop 2) consists of at least one associated RCP providing forced flow for heat transport and an associated SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

(continued)

BASES

APPLICABILITY (continued)

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";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within a 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.

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 12 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If no RCS loop is OPERABLE or in operation, all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of RCS loops are 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. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

This SR requires verification every 12 hours that the secondary side water level in each SG is ≥ 25% wide range. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that the required number of RCPs are OPERABLE ensures that the single failure criterion is met and that an additional RCS loop 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 the required RCPs. 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.

REFERENCES None.

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 transfer of this heat to the Steam Generators (SGs) or Shutdown Cooling (SDC) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either Reactor Coolant Pumps (RCPs) or SDC trains can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RCP or one SDC train for decay heat removal and transport. The flow provided by one RCP loop or SDC train is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal.

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 loops and SDC trains provide this circulation.

RCS Loops – MODE 4 have been identified in 10 CFR. 50.36 (c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two loops or trains, RCS or SDC, be OPERABLE in MODE 4 and one of these loops or trains be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System loops. Any one loop or train in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop or train is required to be OPERABLE to provide redundancy for heat removal.

(continued)

BASES

LCO (continued)

Note 1 permits all RCPs and SDC pumps to be de-energized ≤ 1 hour per 8 hour period. This means that natural circulation should be established, after the operating RCP or SDC pump is secured, using the SGs. Depending on decay heat and current RCS temperature, it may be difficult to establish verifiable natural circulation. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. The intent is to stop any known or direct positive reactivity additions to the RCS due to dilution. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T limits and LTOP) and 10°F subcooling limits; or

b. An alternate heat removal path through the SGs is in operation.

Note 2 requires, that before an RCP may be started with any RCS cold leg temperature ≤ 214°F during cooldown, or ≤ 291°F during heatup, that secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG is < 100°F above each of the RCS cold leg temperatures.

Satisfying the above condition will preclude a large pressure surge in the RCS when the RCP is started.

(continued)

BASES

LCO (continued)

Note 3 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature ≤ 200°F, and no more than 3 RCPs with RCS cold leg temperature >200°F but ≤ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced presure correction factors due to RCP operation (Ref. 1)

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program and has the minimum water level specified in SR 3.4.6.2.

Similarly, for the SDC System, an OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal. RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow, if required.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the SDC 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 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated immediately to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

(continued)

BASES

ACTIONS (continued)

B.1

If only one required SDC train is OPERABLE and in operation, redundancy for heat removal is lost. The plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC train OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC train, it would be safer to initiate that loss from MODE 5 (≤ 210°F) rather than MODE 4 (210°F to 350°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC train operating, in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS loops or SDC trains are OPERABLE, or in operation, all operations involving reduction of RCS boron concentration must be suspended and action to restore one RCS loop or SDC train to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop or train is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one required loop or train is in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) ≥ 25% wide range. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS loop or SDC train 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 the required pumps. 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.

REFERENCES

1. PVNGS Operating License Amendments 52, 38 and 24 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated July 25, 1990.

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 this heat either to the Steam Generator (SG) secondary side coolant or the essential cooling water via the Shutdown Cooling (SDC) heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs 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 side 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.

In MODE 5 with RCS loops filled, the SDC trains are the principal means for decay heat removal. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport. The flow provided by one SDC train is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC train that must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the SGs, each having an adequate water level.

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 SDC trains provide this circulation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Loops – MODE 5 (Loops Filled) have been identified in 10 CFR 50.36 (c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require at least one of the SDC trains be OPERABLE and in operation with an additional SDC train OPERABLE or secondary side water level of each SG shall be ≥ 25% wide range level. One SDC train provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC train is normally maintained OPERABLE as a backup to the operating SDC train to provide redundant paths for decay heat removal. However, if the standby SDC train is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ 25% wide range. Should the operating SDC train fail, the SGs could be used to remove the decay heat.

Note 1 permits all SDC pumps to be de-energized ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC trains are to be limited to situations where pressure and temperature increases can be maintained well within the allowable pressure (pressure and temperature and low temperature overpressure protection) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

This LCO is modified by a Note that prohibits boron dilution when SDC forced flow is stopped because an even concentration distribution cannot be ensured. The intent is to stop any known or direct positive reactivity changes to the RCS due to dilution. Core outlet temperature is to be maintained at least 10°F below saturation temperature, so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as the backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

(continued)

BASES

LCO (continued)

In MODE 5, it is sometimes necessary to stop all RCP or SDC forced circulation. This is permitted to change operation from one SDC train to the other, perform surveillance or startup testing, perform the transition to and from the SDC, or to avoid operation below the RCP minimum net positive suction head limit. The time period is acceptable because natural circulation is acceptable for decay heat removal the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC train to be inoperable for a period of up to 2 hours provided that the other SDC train is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable train during the only time when such testing is safe and possible.

Note 3 requires that before an RCP may be started with any RCS cold leg temperature $\leq$ 214°F during a cooldown, or $\leq$ 291°F during a heatup, the secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG must be < 100°F above each of the RCS cold leg temperatures.

Satisfying the above condition will preclude a low temperature overpressure event due to a thermal transient when the RCP is started.

Note 4 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature $\leq$ 200°F, and no more than 3 RCPs with RCS cold leg temperature > 200°F but $\leq$ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced pressure correction factors due to RCP operation (Ref. 3).

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of SDC trains from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal.

(continued)

BASES

LCO (continued)

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow (current Section XI), if required. An OPERABLE SG can perform as a heat sink when it is OPERABLE in accordance with the SG Tube Surveillance Program and has the minimum water level specified in SR 3.4.7.2.

When entering RCS Loops-MODE 5 Loops Filled from RCS Loops-MODE 5 Loops not Filled the additional requirement of total gas concentration must be addressed for SGs to be considered as a heat sink. A total gas concentration of < 20 cc/kg is required for MODE 5 operations. This limit ensures that gases coming out of solution in the SG U-tubes will not adversely affect natural circulation with RCS pressure at atmospheric conditions. Normal operating procedures implement the findings for determination of when RCS loops are considered filled, which in turn allows for transition from RCS Loops-MODE 5 Loops not Filled to RCS Loops-MODE 5 Loops Filled.

The ability to feed and the ability to steam SGs is not a requirement. Since RCS temperature is less than 210°F in MODE 5-Loops Filled, no boiling (i.e. loss of SG inventory) will occur, therefore, the ability to feed and the ability to steam SGs is not a requirement. However, a means to feed and steam the SGs, whenever the Unit is in MODE 5-Loops Filled, should be provided. Feed sources available are not limited only to Essential Auxiliary Feedwater Pumps and the Condensate Storage Tank. Steaming capability is usually via ADVs. Also, the RCS must be intact (ability to pressurize to at least 100 psia) for the SGs to be considered as a heat sink. With the RCS not intact a majority of heat removal, assuming a loss of SDC flow, will be out the vent (Ref. 2). Therefore, with the RCS not intact, transition to LCO 3.4.8, RCS Loops-MODE 5 Loops not Filled, is appropriate.

(continued)

BASES (continued)

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation to remove decay heat from the core and to provide proper boron mixing. One SDC train provides sufficient circulation for these purposes.

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 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If a SDC train is inoperable and any SGs have secondary side water levels < 25% wide range, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC train to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

If the required SDC train is not OPERABLE or no SDC train is in operation, all operations involving the reduction of RCS boron concentration must be suspended. Action to restore one SDC train to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that one SDC train is in operation and circulating reactor coolant at a flow rate of greater than or equal to 3780 gpm. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the standby SDC train should the operating train be lost.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ 25% wide range level ensures that redundant heat removal paths are available if the second SDC train is inoperable. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC trains are OPERABLE, this SR is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.7.3

Verification that the second SDC train is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional train 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 the required pumps. The Surveillance is required to be performed when the LCO requirement is being met by one of two SDC trains, e.g., both SGs have < 25% wide range water level. 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.

(continued)

BASES (continued)

REFERENCES

1. Not Used

2. CEN-PSD-770 Analysis for Lower Mode Functional Recovery Guidelines.

3. PVNGS Operating License Amendmants 52, 38, and 24 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated July 25, 1990.

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 and transfer of this heat to the Shutdown Cooling (SDC) 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 the SDC System can be used for coolant circulation. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in determining the time available for mitigation of the accidental boron dilution event. The SDC trains provide this circulation. The flow provided by one SDC train is adequate for decay heat removal and for boron mixing.

RCS loops–MODE 5 (loops not filled) have been identified in 10 CFR 50.36 (c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require a minimum of two SDC trains be OPERABLE and one of these trains be in operation. An OPERABLE train is one that is capable of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the SDC System unless forced flow is used. A minimum of one running SDC pump meets the LCO requirement for one train in operation. An additional SDC train is required to be OPERABLE to meet the single failure criterion.

(continued)

BASES

LCO (continued)

Note 1 permits all SDC pumps to be de-energized ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when SDC forced flow is stopped.

Note 2 allows one SDC train to be inoperable for a period of 2 hours provided that the other train is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable train during the only time when these tests are safe and possible.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow (current Section XI), if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the SDC 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";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If a SDC train is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second train to OPERABLE status. The Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If no SDC train is OPERABLE or in operation, except as provided in NOTE 1, all operations involving the reduction of RCS boron concentration must be suspended. Action to restore one SDC train to OPERABLE status and operation must be initiated immediately. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one SDC train is in operation and circulating reactor coolant. Verification includes flow rate of greater than or equal to 3780 gpm, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions.

SR 3.4.8.2

Verification that the required number of trains are OPERABLE ensures that redundant paths for heat removal are available and that an additional train 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 indicated power available to the required pumps. 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.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEMS (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 backup heater controls, and emergency power supplies. Pressurizer safety valves and pressurizer vents are addressed by LCO 3.4.10 "Pressurizer Safety Valves-MODES 1,2, and 3," LCO 3.4.11 "Pressurizer Safety Valves-MODE 4," and LCO 3.4.12 "Pressurizer Vents", respectively.

The maximum steady state water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and proper pressure response for anticipated design basis transients. The maximum and minimum steady state water level limit serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus in the preferred state for heat transport; and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum steady state water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices

(continued)

BASES

BACKGROUND (continued)

(pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the Safety Limit of 2750 psia.

The minimum steady state water level in the pressurizer assures pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered.

The requirement to have two groups of pressurizer heaters ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. 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 noncondensable gases normally present.

Safety analyses presented in the UFSAR 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 1), is the reason for their inclusion. The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to keep the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses. The pressurizer satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level ≥ 27% and ≤ 56% ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 125 kW and capable of being powered from an emergency power supply. The minimum heater capacity required is sufficient to 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 subcooling margin to saturation can be obtained in the loops.

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. It is assumed pressurizer level is under steady state conditions. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational

(continued)

BASES

APPLICABILITY (continued)

perturbation, such as reactor coolant pump startup. The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) System," applies. The LCO does not apply to MODES 5 and 6 with partial loop operation. Also, a Note has been added to indicate the limit on pressurizer level may be exceeded during short term operational transients such as a THERMAL POWER ramp increase of > 5% RTP per minute or a THERMAL POWER step increase of > 10% RTP.

In MODES 1, 2, and 3, there is the 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 gives the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODES 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Shutdown Cooling System is in service and therefore the LCO is not applicable.

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses.

Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further pressure and temperature reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

(continued)

BASES

ACTIONS (continued)

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 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 heaters.

C.1 and C.2

If one required group of pressurizer heaters is inoperable and cannot be restored within 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 Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance ensures that during steady state operation, pressurizer water 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. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.9.2

The Surveillance 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 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the four spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System, four valves are used to ensure that the Safety Limit (SL) of 2750 psia is not exceeded for analyzed transients during operation in MODES 1, 2 and 3. One safety valve used for MODE 4. For MODE 5, and MODE 6 with the head on, overpressure protection is provided by operating procedures and the LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME, Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2475 psia +3%, -1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. 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 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of four pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of four safety valves and assumes that the valves open at the high range of the setting (2475 psia + 3%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, loss of main feedwater, or main feedwater line break accident. The Loss of Load with Delayed Reactor Trip accident establishes the minimum safety valve capacity. The Loss of Load with Delayed Reactor Trip accident is assumed to occur at 100% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The four pressurizer safety valves are set to open at 25 psia less than RCS design pressure (2475 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. 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.

APPLICABILITY

In MODES 1, 2, and 3, OPERABILITY of four 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 conservatively included, although the listed accidents may not require four safety valves for protection.

(continued)

BASES

APPLICABILITY (continued)

The requirements for overpressure protection in other MODES are covered by LCO 3.4.11, "Pressurizer Safety Valves-MODE 4," and LCO 3.4.13, "LTOP System."

The Note allows entry into MODES 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 72 hour exception is based on 18 hour outage time for each of the four valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

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 RCPB.

B.1 and B.2

If the Required Action 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 within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 four pressurizer safety valves.

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 Section XI of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3%, - 1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift (Ref. 2). The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, Section XI.

2. PVNGS Operating License Amendment Nos. 75, 61, and 47 for Units 1, 2, and 3, respectively, and associated NRC Safety Evaluation dated May 16, 1994.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Safety Valves-MODE 4

BASES

BACKGROUND

The purpose of the four spring loaded pressurizer safety valves is to provide RCS overpressure protection. One safety valve is used for portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the head on, overpressure protection is provided by operating procedures and the LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME, Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2475 psia +3%, -1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. 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 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.

(continued)

BASES

BACKGROUND (continued)

Pressurizer Safety Valve Requirements

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit (SL) of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam at valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown above L-Top System temperatures.

Shutdown Cooling System Suction Line Relief Valve Requirements

A single Shutdown Cooling System suction line relief valve provides overpressure relief capability and will prevent RCS overpressurization in the event that no pressurizer safety valves are OPERABLE.

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of four pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of four safety valves and assumes that the valves open at the high range of the setting (2475 psia + 3%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, loss of main feedwater, or main feedwater line break accident. The Loss of Load with Delayed Reactor Trip accident establishes the minimum safety valve capacity. The Loss of Load with Delayed Reactor Trip accident is assumed to occur at 100% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

One pressurizer safety valve is required to be OPERABLE in MODE 4 with no Shutdown Cooling System suction line relief valves in service. The four pressurizer safety valves are set to open 25 psia less than RCS design pressure (2475 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL to maintain accident analysis assumptions, and to comply with ASME Code requirements. The limit protected by this specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of all 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.

APPLICABILITY

In MODE 4 above the LTOP System temperatures OPERABILITY of one valve is required. MODE 4 is conservatively included, although the listed accidents may not require a safety valve for protection.

The requirements for overpressure protection in other MODES and in MODE 4 at or below the LTOP System temperatures are covered by LCOs 3.4.10, "Pressurizer Safety Valves - MODES 1, 2 and 3," and LCO 3.4.13, LTOP System.

The Note allows entry into MODES 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 72 hour exception is based on 18 hour outage time for each of the four valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, and A.3

If all pressurizer safety valves are inoperable, the plant must be brought to a condition where overpressure protection is provided, then to a MODE in which the requirement does not apply. To achieve this status, one Shutdown Cooling System suction line relief must be placed in service immediately, then the plant must be brought to at least MODE 4 with any RCS cold leg temperatures ≤ 214°F during cooldown or ≤ 291°F during heatup within 8 hours, so that LCO 3.4.13 (LTOP System) would apply. It is reasonable to pursue the ACTION to place a shutdown cooling system suction relief valve in service immediately (without delay) because the plant is already within the shutdown cooling system entry temperature of less than 350°F. The Completion Time of immediately requires that the required action be pursued without delay and in a controlled manner, and reflects the importance of maintaining the RCS overprotection system. The 8 hours allowed to be in MODE 4 with any RCS temperatures ≤ 214°F during cooldown or ≤ 291°F during heatup is reasonable, based on operating experience, to reach this condition without challenging plant systems.

For the Shutdown Cooling System suction line relief valve that is required to be in service in accordance with Required Action A.1, SR 3.4.11.2 and SR 3.4.11.3 must be performed or verified performed within 12 hours. This ensures that the required Shutdown Cooling System suction line relief valve is OPERABLE. A Shutdown Cooling System suction line relief valve is OPERABLE when its isolation valves are open, its lift setpoint is set at 467 psig or less, and testing has proven its ability to open at that setpoint.

If the Required Actions and associated Completion Times are not met, overpressurization is possible.

The 8 hours Completion Time to be in MODE 4 with any RCS cold leg temperatures ≤214°F during cooldown or ≤291°F during heatup places the unit in a condition where the LCO does not apply.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3%, -1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift (Ref. 3). The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SR 3.4.11.2

SR 3.4.11.2 requires that the required Shutdown Cooling System suction line relief valve is OPERABLE by verifying its open pathway condition either:

a. Once every 12 hours for a valve that is unlocked, not sealed, or otherwise not secured open in the vent pathway, or

b. Once every 31 days for a valve that is locked, sealed or otherwise secured open in the vent pathway.

The SR has been modified by a Note that requires performance only if a Shutdown Cooling System suction line relief valve is being used for overpressure protection. The Frequencies consider operating experience with mispositioning of unlocked and locked pathway vent valves.

SR 3.4.11.3

SRs are specified in the Inservice Testing Program. Shutdown Cooling System suction line relief valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. The Shutdown Cooling System suction line relief valve setpoint is 467 psig.

(continued)

BASES (continued)

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, Section XI.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

3. PVNGS Operating License Amendment Nos. 75, 61, and 47 for Units 1, 2, and 3 respectively, and associated NRC Safety Evaluation dated May 16, 1994.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Pressurizer Vents

BASES

BACKGROUND

The pressurizer vent is part of the reactor coolant gas vent system (RCGVS) as described in UFSAR 18.II.B.1 (Ref.1). The pressurizer can be vented remotely from the control room through the following four paths (see UFSAR Figure 18.II.B-1):

1. From the pressurizer vent through SOV HV-103, then through SOV HV-105 to the reactor drain tank (RDT).

2. From the pressurizer vent through SOV HV-103, then through SOV HV-106 directly to the containment atmosphere.

3. From the pressurizer vent through SOVs HV-108 and HV-109, then through SOV HV-105 to the reactor drain tank (RTD).

4. From the pressurizer vent through SOVs HV-108 and HV-109, then through SOV HV-106 directly to the containment atmosphere.

The RCGVS also includes the reactor head vent, which can be used along with the pressurizer vent to remotely vent gases that could inhibit natural circulation core cooling during post accident situations. However, this function does not meet the criteria of 10 CFR 50.36(c)(2)(ii) to require a Technical Specification LCO, and therefore the reactor head vent is not included in these Technical Specifications.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The requirement for pressurizer path vent path to be OPERABLE is based on the steam generator tube rupture (SGTR) with loss of offsite power (LOP) and a single failure safety analysis, as described in UFSAR 15.6.3 (Ref. 4). It is assumed that the auxiliary pressurizer spray system (APSS) is not available for this event. Instead, RCS depressurization is performed, 2 hours after the initial SGTR, by venting the RCS via a pressurizer vent path and throttling HPSI flow. The analysis also incorporates an additional failure by assuming that only the smallest of the four available pressurizer vent paths is used. This is identified as the orificed flow path to the RDT.

The results of the analysis for steam generator tube rupture with a loss of offsite power and a fully stuck open ADV using the pressurizer vent system, forwarded to the NRC in Reference 3, states that the analysis assumes that the APSS is inoperable and the pressurizer gas vent system performs the functions of RCS depressurization. The staff has reviewed and accepted the results of the analysis and the design of the pressurizer gas vent system. The staff's detailed evaluation has been reported in Supplement No. 9 to PVNGS SER (Ref. 2).

The pressurizer vent paths satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requires four pressurizer vent paths be OPERABLE. The four vent paths are:

1. From the pressurizer vent through SOV HV-103, then through SOV HV-105 to the reactor drain tank (RTD).
2. From the pressurizer vent through SOV HV-103, then through SOV HV-106 directly to the containment atmosphere.
3. From the pressurizer vent through SOVs HV-108 and HV-109, then through SOV HV-105 to the reactor drain tank (RTD).
4. From the pressurizer vent through SOVs HV-108 and HV-109, then through SOV HV-106 directly to the containment atmosphere.

(continued)

BASES

LCO (continued)

A vent path is flow capability from the pressurizer to the RDT or from the pressurizer to containment atmosphere. Loss of any single valve in the pressurizer vent system will cause two flow paths to become inoperable. A pressurizer vent path is required to depressurize the RCS in a SGTR design basis event which assumes LOP and APSS unavailable.

APPLICABILITY

In MODES 1, 2, 3, and MODE 4 with RCS pressure $\geq$ 385 psia the four pressurizer vent paths are required to be OPERABLE. The safety analysis for the SGTR with LOP and a Single Failure (loss of APSS) credits a pressurizer vent path to reduce RCS pressure.

In MODES 1, 2, 3, and MODE 4 with RCS pressure $\geq$ 385 psia the SGs are the primary means of heat removal in the RCS, until shutdown cooling can be initiated. In MODES 1, 2, 3, and MODE 4 with RCS pressure $\geq$ 385 psia, assuming the APSS is not available, the pressurizer vent paths are the credited means to depressurize the RCS to Shutdown Cooling System entry conditions. Further depressurization into MODE 5 requires use of the pressurizer vent paths. In MODE 5 with the reactor vessel head in place, temperature requirements of MODE 5 (< 210°F) ensure the RCS remains depressurized. In MODE 6 the RCS is depressurized.

ACTIONS

A.1

If two or three pressurizer vent paths are inoperable, they must be restored to OPERABLE status. Loss of any single valve in the pressurizer vent system will cause two flow paths to become inoperable. Any vent path that provides flow capability from the pressurizer to the RDT or to the containment atmosphere, independent of which train is powering the valves in the flow path, can be considered an operable vent path. The Completion Time of 72 hours is reasonable because there is at least one pressurizer vent path that remains OPERABLE.

(continued)

BASES

B.1

If all pressurizer vent paths are inoperable, then restore at least one pressurizer vent path to OPERABLE status. The Completion Time of 6 hours is reasonable to allow time to correct the situation, yet emphasize the importance of restoring at least one pressurizer vent path. If at least one pressurizer vent path is not restored to OPERABLE within the Completion Time, then Action C is entered.

C.1

If the required Actions, A and B, cannot be met within the associated Completion Times, 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 RCS pressure < 385 psia within 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 without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

SR 3.4.12.1 requires complete cycling of each pressurizer vent path valve. The vent valves must be cycled from the control room to demonstrate their operability. Pressurizer vent path valve cycling demonstrates its function. The frequency of 18 months is based on a typical refueling cycle and industry accepted practice. This surveillance test must be performed in Mode 5 or Mode 6.

SR 3.4.12.2

SR 3.4.12.2 requires verification of flow through each pressurizer vent path. Verification of pressurizer vent path flow demonstrates its function. The frequency of 18 months is based on a typical refueling cycle and industry accepted practice. This surveillance test must be performed in Mode 5 or Mode 6.

(continued)

BASES

REFERENCES

1. UFSAR, Section 18.

2. NUREG-0857, initial issue, November 1981, through Supplement 12, November 1987.

3 Letter from Arizona Nuclear Power Project to NRC (ANPP-33905) dated November 4, 1985, "Information Concerning the PVNGS Auxiliary Pressurizer Spray."

4. UFSAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 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. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. 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 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having adequate pressure relief capacity. The pressure relief capacity requires either two OPERABLE redundant Shutdown Cooling System suction line relief valves or the RCS depressurized and an RCS vent of sufficient size. One Shutdown Cooling System suction line relief valve or the RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

BACKGROUND (continued)

The LTOP System for pressure relief consists of two Shutdown Cooling System suction line relief valves or an RCS vent of sufficient size. Two relief valves are required for redundancy. One Shutdown Cooling System suction line relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability.

Shutdown Cooling System Suction Line Relief Valve Requirements

As designed for the LTOP System, each Shutdown Cooling System suction line relief valve is designed to lift and relieve RCS pressure if RCS pressure approaches the Shutdown Cooling System suction line relief valve lift setpoint.

Each Shutdown Cooling System suction line relief valve is designed to protect the reactor vessel given a single failure in addition to a failure that initiated the pressure transient. No single failure of a Shutdown Cooling System suction line relief valve isolation valve (SI-651, 652, 653, or 654) will prevent one Shutdown Cooling System suction line relief valve from performing its intended function (Ref. 7).

The OPERABILITY of two Shutdown Cooling System suction line relief valves, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from analyzed pressure transients. Either Shutdown Cooling System suction line relief valve provides overpressure protection for the RCS due to the most limiting transients initiated by a single operator or equipment failure.

a. The start of an idle RCP with secondary water temperature of the SG ≤ 100°F above RCS cold leg temperatures

b. An inadvertent SIAS with two HPSI pumps injecting into a water solid RCS, three charging pumps injecting, and letdown isolated.

These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures (Refs. 7, 8 and 9).

(continued)

BACKGROUND (continued)

Shutdown Cooling System Suction Line Relief Valve Requirements. (continued)

When a Shutdown Cooling System suction line relief valve lifts due to an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the Shutdown Cooling System suction line relief valve releases coolant, the system pressure decreases until valve reseat pressure is reached and the Shutdown Cooling system suction line relief valve closes.

At low temperatures with the Shutdown Cooling System suction line relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that P-T limits are not exceeded. These P-T limits are usually applicable to a finite time period such a one cycle, 5 EFPY, etc. and are based upon irradiation damage prediction by the end of the period. Accordingly, each time P-T limits change, the LTOP System needs to be reanalyzed and modified, if necessary, to continue its function.

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 specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with any RCS cold leg temperature exceeding 214°F during cooldown or 291°F during heatup, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At about 214°F and below, during cooldown or 291°F and below during heatup, overpressure prevention falls to the OPERABLE Shutdown Cooling System suction line relief valves or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

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 P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the Shutdown Cooling System suction line relief valve method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

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 shutdown cooling (SDC); or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

References 3, 7, 8 and 9 analyses demonstrate that either one Shutdown Cooling System suction line relief valve or the RCS vent can maintain RCS pressure below limits for the two most limiting analyzed events:

a. The start of an idle RCP with secondary water temperature of the SG ≤ 100°F above RCS cold leg temperatures.

b. An inadvertent SIAS with two HPSI pumps injecting into a water solid RCS, three charging pumps injecting, and letdown isolated.

Fracture mechanics analyses established the temperature of LTOP Applicability at 214°F and below during cooldown and 291°F and below during heatup. Above these temperatures, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 32 effective full power years of operation.

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 and 5).

The fracture mechanics analyses show that the vessel is protected when the Shutdown Cooling System suction line relief valves are set to open at or below 467 psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient. The Shutdown Cooling System suction line relief valves setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The Shutdown Cooling System suction line relief valves setpoints will be re-evaluated for compliance 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 embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of 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.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The Shutdown Cooling System suction line relief valves are considered active components. Thus, the failure of one Shutdown Cooling System suction line relief valve represents the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 16 square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve.

The RCS vent size will also be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP System satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the 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.

The elements of the LCO that provide overpressure mitigation through pressure relief are:

a. Two OPERABLE Shutdown Cooling System suction line relief valves; or

b. The depressurized RCS and an RCS vent.

A Shutdown Cooling System suction line relief valve is OPERABLE for LTOP when its isolation valves are open, its lift setpoint is set at 467 psig or less and testing has proven its ability to open at that setpoint.

An RCS vent is OPERABLE when open with an area $\geq$ 16 square inches. For an RCS vent to meet the specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

LCO (continued)

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The Note requires that, before an RCP may be started, the secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG is ≤ 100°F above each of the RCS cold leg temperatures. Satisfying this condition will preclude a large pressure surge in the RCS when the RCP is started.

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is ≤ 214°F during cooldown or ≤ 291°F during heatup, 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 214°F during cooldown and 291°F during heatup. The requirements for overpressure protection in MODES 1, 2 and 3, and in MODE 4 above the LTOP System temperatures are covered by LCO 3.4.10, "Pressurizer Safety Valves - MODES 1, 2, and 3," and LCO 3.4.11, "Pressurizer Safety Valves - MODE 4." When the reactor vessel head is off overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.

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.

The Applicability is modified by a Note stating when one or more cold legs reach 214°F, this LCO remains applicable during periods of steady state temperature conditions until all RCS cold leg temperatures reach 291°F. Also, if a cooldown is terminated prior to reaching 214°F and a heatup is commenced, this LCO is applicable until all RCS cold leg temperatures reach 291°F. This Note provides clarification about Applicability intent. Since PVNGS uses two different temperatures at which the Shutdown Cooling System suction line relief valves must be placed in service there is some possibility of confusion. This Note clarifies those circumstances where the Shutdown Cooling System suction line relief valves must be placed in service.

(continued)

BASES (continued)

ACTIONS

A.1

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable.

In MODE 4 when any RCS cold leg temperature is ≤ 214°F during cooldown or ≤ 291°F during heatup with one Shutdown Cooling System suction line relief valve inoperable, two Shutdown Cooling System suction line relief valves must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on the facts that only one Shutdown Cooling System suction line relief valve 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.

B.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required Shutdown Cooling System suction line relief valve inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore inoperable valve to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore two Shutdown Cooling System suction line relief valves OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is a reasonable amount of time to investigate and repair several types of Shutdown Cooling System suction line relief valve failures without exposure to a lengthy period with only one Shutdown Cooling System suction line relief valve OPERABLE to protect against overpressure events.

(continued)

ACTIONS (continued)

C.1

If two required Shutdown Cooling System suction line relief valves are inoperable, or if a Required Action and the associated Completion Time of Condition A or B are not met, the RCS must be depressurized and a vent established within 8 hours. The vent must be sized at least 16 square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel. For personnel safety considerations, the RCS cold leg temperature must be reduced to less than 200°F prior to venting.

The Completion Time of 8 hours to depressurize and vent the RCS is based on 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 3.4.13.1 and 3.4.13.2

SR 3.4.13.1 and SR 3.4.13.2 require verifying that the RCS vent is open ≥ 16 square inches or that the Shutdown Cooling System suction line relief valves be aligned to provide overpressure protection for the RCS is proven OPERABLE by verifying its open pathway condition either:

Shutdown Cooling System suction/line relief valves

a. Once every 12 hours for a valve that is unlocked, not sealed, or otherwise not secured open in the vent pathway, or

b. Once every 31 days for a valve that is locked, sealed, or otherwise secured open in the vent pathway.

RCS Vent

a. Once every 12 hours for a vent pathway that is unlocked, not sealed, or otherwise not secured open

b. Once every 31 days for a vent pathway that is locked, sealed, or otherwise secured open.

For an RCS vent to meet the specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 and 3.4.13.2 (continued)

The passive vent arrangement must only be open (vent pathway exists) to be OPERABLE. These Surveillances need only be performed if the vent or the Shutdown Cooling System suction line relief valves are being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked pathway vent valves, and passive pathway obstructions.

SR 3.4.13.3

SRs are specified in the Inservice Testing Program. Shutdown Cooling System suction line relief valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 10), which provides the activities and the Frequency necessary to satisfy the SRs. The Shutdown Cooling System suction line relief valve set point is 467 psig.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 15.
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.
7. UFSAR, Section 5.2.
8. V-PSAC-009, Pressure Transient Analysis.
9. V-PSAC-010, Mass Input Pressure Transient in Water Solid RCS.
10. ASME, Boiler and Pressure Vessel Code, Section XI.
11. 13-COO-93-016, Sensitivity Study on Pressurizer Vent Paths vs. Days Post Shutdown.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 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 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 analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA).

(continued)

BASES (continued)

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 assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

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 UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits).

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.

(continued)

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 identified 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.

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 leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident analysis. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

(continued)

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The 720 gallon per day limit on primary to secondary LEAKAGE through any one SG allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.

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.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary 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 if unidentified, identified, or primary to secondary 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. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required 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.

C.1

If one or more SGs are inoperable, due to SR 3.4.14.2, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation near operating pressure have elapsed. This means that once steady state operating conditions are established, 12 hours is allowed for completing the Surveillance if the Surveillance Frequency interval was exceeded in MODE 5 or 6. Further discussion of SR note format is found in Section 1.4, Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

The Note in the Frequency column allows for SR 3.4.14.1 nonperformance due to planned or unplanned power manipulations. This Note is not intended to allow power manipulations solely for the purpose of avoiding SR 3.4.14.1 performance. Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. 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. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.14.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 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 RCS pressure boundary that 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 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 PIVs in series in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.14, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two valves in series is determined by a water inventory balance (SR 3.4.14.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage 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 condition 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.

(continued)

BASES

BACKGROUND (continued)

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Shutdown Cooling (SDC) System; and

b. Safety Injection System;

The PIVs are listed in UFSAR section (Ref. 6).

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 SDC System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the Reactor Coolant Pressure Boundary (RCPB), and the subsequent pressurization of the SDC System downstream of the PIVs from the RCS. Because the low pressure portion of the SDC System is typically designed for 485 psig, overpressurization failure of the SDC 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

(continued)

BASES

LCO (continued)

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.

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.

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 SDC flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the SDC 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 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based on 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

(continued)

BASES

ACTIONS (continued)

have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flowpath must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as PIVs and must be in the RCPB.

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 if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.

B.1 and B.2

If leakage cannot be reduced the system isolated or other Required Actions accomplished, 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 5 within 36 hours. This Action reduces 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 or 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

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 up to 18 months, a typical refueling cycle, 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), is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform the Surveillance under conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

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.

The SDC PIVs excepted in two of the three FREQUENCIES are UV-651, UV-652, UV-653, and UV-654, due to position indication of the valves in the control room.

Although not explicitly required by SR 3.4.15.1, performance of leakage testing to verify leakage is below the specified limit must be performed prior to returning a valve to service following maintenance, repair or replacement work on the valve in order to demonstrate operability.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

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 complimentary 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 SDC System when the SDC System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the SDC shutdown cooling flow path must be leakage rate tested after SDC is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.15.2

Verifying that the SDC open permissive interlocks are OPERABLE ensures that RCS pressure will not pressurize the SDC system beyond 125% of its design pressure of 485 psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be 410 psia to open the valves. This setpoint ensures the SDC design pressure will not be exceeded and the SDC relief valves will not lift. The 18 month Frequency is based on the need to perform this 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.

(continued)

BASES (continued)

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. UFSAR, Section 3.9.6.2
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 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 (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.

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump monitor consists of instrumentation used to monitor containment sump level and flow (pump run time). The containment sump used to collect unidentified LEAKAGE is instrumented to alarm at 1.0 gpm above normal flow for 1 hour (Ref. 3). This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. 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. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and responses to RCS LEAKAGE.

(continued)

BASES

BACKGROUND (continued)

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. A 1°F increase in dew point is well within the sensitivity range of available instruments.

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 sump. 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.

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 are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing a sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

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 RCS leakage detection instrumentation is described in the UFSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to help identify the location of the LEAKAGE source.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 are necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should leakage occur detrimental to the safety of the facility and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR (C)(2)(ii).

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor in combination with a particulate and gaseous radioactivity monitor (RU-1) provides an acceptable minimum. It has been determined that it is acceptable to continue to call the containment sump OPERABLE with one containment sump pump out of service.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is $\leq$ 210°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 is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

The Actions are modified by a Note that indicates the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump and required containment atmosphere radioactivity monitor channels are inoperable. This allowance is provided because other means are available to monitor for RCS LEAKAGE.

A.1 and A.2

If the containment sump monitor is inoperable, no other form of sampling can provide the equivalent information.

However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.14.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days 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.

B.1.1, B.1.2, and B.2

With either the gaseous or particulate containment atmosphere radioactivity monitoring instrumentation channels 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 3.4.14.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of both of the radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

(continued)

BASES (continued)

ACTIONS (continued)

C.1

If any Required Action of Condition A or B cannot be met within the required 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.

D.1

If all required monitors are inoperable, no automatic means of monitoring leakage are available and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.16.2

SR 3.4.16.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. The test ensures that the monitor can perform its function in the desired manner. 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 it proper for detecting degradation.

The alarm setpoints for the containment building atmosphere monitor (RU-1) are:

particulate	$\leq 2.3 \times 10^{-6}$ μCi/cc CS-137
gaseous	$\leq 6.6 \times 10^{-2}$ μCi/cc Xe-133

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3, SR 3.4.16.4

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. Operating experience has shown this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. UFSAR, Section 5.2.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1) specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. 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 an 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 ensure 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 an SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limits and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.

The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam 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 3.4.17-1 for more than 48 hours. The above-limit permissible iodine levels shown in Figure 3.4.17-1 are acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The allowable limits shown on Figure 3.4.17-1 accomodate possible iodine spiking phenomenon which may occur following changes in thermal power.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (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.

(continued)

BASES

LCO (continued)

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.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS cold leg temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity is necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS cold leg temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and 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 the limits of Figure 3.4.17-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample.

Sampling must continue for trending. 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.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception 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.

B.1

If a Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.17-1, the reactor must be brought to MODE 3 with RCS cold leg temperature < 500°F within 6 hours. The allowed Completion Time of 6 hours is required to reach MODE 3 below 500°F without challenging plant systems.

C.1 and C.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS cold leg temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and minimizes the potential for venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

The Surveillance requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. 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 RCS cold leg temperature at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.17.2

This Surveillance is performed to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. During normal operation, the 14 day Frequency is adequate to trend changes in the iodine activity level considering gross activity is monitored every 7 days. The 14 day surveillance frequency is modified by the Note "Only required to be performed in MODE 1." This is acceptable because the level of fission products generated in MODES 2 and 3 is much less than in MODE 1. The Frequency, between 2 hours and 6 hours after a power change of ≥ 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. One sample is sufficient if the plant has gone through a shutdown or if the transient is complete in 6 hours.

SR 3.4.17.2 14 day Frequency is modified by a Note which requires the Surveillance to only be performed in MODE 1. This is required because the level of fission products generated in other MODES is much less. Also, fuel failures associated with fast power changes is more apt to occur in MODE 1 than in MODES 2 or 3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ 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 $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours should the 184 day Frequency interval be exceeded. Further discussion of SR Note format is found in Section 1.4, Frequency. This ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. UFSAR, Section 15.6.3.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs) - Operating

BASES

BACKGROUND

The functions of the four SITs 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.

The refill phase of a LOCA follows immediately where 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 the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer 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.

The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The SITs are passive components, since no operator or control action is required for them to perform their function. Internal tank pressure is sufficient to discharge the contents to the RCS, if RCS pressure decreases below the SIT pressure.

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) Systems. Each SIT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

(continued)

BASES

BACKGROUND (continued)

Additionally, the isolation valves are interlocked with the pressurizer pressure instrumentation channels to ensure that the valves will automatically open as RCS pressure increases above SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps and HPSI pumps cannot deliver flow until the Diesel Generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

During this event, the SITs discharge to the RCS as soon as RCS pressure decreases to below SIT pressure. As a conservative estimate, no credit is taken for SI pump flow until the SITs are empty. This results in a minimum effective delay of over 55 seconds, during which the SITs must provide the core cooling function. The actual delay time does not exceed 29 seconds. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA also assumes 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 SITs, with pumped flow then providing continued cooling. As break size decreases, the SITs and HPSI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the SITs continues to decrease until they are not required, and the HPSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 3) for the ECCS, 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 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. The core is maintained in a coolable geometry.

Since the SITs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves and SIT nitrogen vent valves, however, are not single failure proof;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

therefore, whenever the SIT motor operated isolation valves are open, power is removed from their operators and the switch is key locked open. Whenever the SIT vent valves are closed, power is removed with a keylock switch.

These precautions ensure that the SITs are available during an accident (Ref. 4). With power supplied to the valves, a single active failure could result in a valve failure, which would render one SIT unavailable for injection. If a second SIT is lost through the break, only two SITs would reach the core. Active failures that could affect the SITs would be the closure of a motor operated outlet valve or opening of a solenoid operated nitrogen vent valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum amount of boron inventory in the SITs.

A minimum of 1750 cubic feet of borated water, and a maximum of 1950 cubic feet of borated water are used in the safety analyses as the volume in the SITs. To allow for instrument inaccuracy, a 28% narrow range (corresponding to 1802 cubic feet) and a 72% narrow range (corresponding to 1914 cubic feet) are specified. The analyses are based upon the cubic feet requirements; the percentage figures are provided in the LCO for operator use because the level indicator provided in the control room is marked in percentages, not in cubic feet.

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

A minimum pressure of 588 psig and a maximum pressure of 637 psig are used in the analyses. To allow for instrument accuracy, a 600 psig minimum and 625 psig maximum are specified. The maximum allowable boron concentration of 4400 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The 2300 ppm minimum boron concentration in the SITs assures that the back leakage from the RCS will not dilute the SITs below the minimum boron concentration in the safety analysis. The minimum safety analysis boron requirements of 2000 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all Control Element Assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the Refueling Water Tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS during a large break LOCA.

The SITs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. Four SITs are required to be OPERABLE to ensure that 100% of the contents of three of the SITs will reach the core during a LOCA.

(continued)

BASES

LCO (continued)

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For a SIT to be considered OPERABLE, the motor operated isolation valve must be fully open, power removed and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and MODES 3 and 4 with pressurizer pressure ≥ 1837 psia, the SIT OPERABILITY requirements are based on an assumption of full power operation. Although cooling requirements decrease as power decreases, the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

The SIT functional requirements in MODES 3 and 4 with pressurizer pressure < 1837 psia are described in LCO 3.5.2, "SIT - Shutdown".

In MODE 4 with pressurizer pressure < 430 psia, the SIT motor operated isolation valves may be closed to isolate the SITs from the RCS but must remain energized. This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs. In this situation, manual actions would be required to open the SIT motor operated isolation valves (i.e., a manually initiated SIAS).

In MODES 5 and 6, the SITs are not required and the SIT motor operated isolation valves are closed as required to isolate the SITs from the RCS.

(continued)

BASES

ACTIONS

A.1

If the boron concentration of one SIT is not within limits, the SIT must be returned to OPERABLE status within 72 hours. If the boron concentration is not within limits, ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if a SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one SIT is inoperable for a reason other than boron concentration, the SIT must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three SITs 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 1 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 SIT to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

ACTIONS

C.1 and C.2

If the SIT cannot be restored 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 pressurizer pressure reduced to < 1837 psia 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.

(continued)

BASES

ACTIONS

C.1 and C.2 (Continued)

Specification 3.5.2, "SITs - Shutdown", further requires the plant to be in Mode 5 within 24 hours if the SIT inoperability was discovered but not restored while in the applicability of Specification 3.5.1, "SITs - Operating".

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each SIT isolation valve is fully open, as indicated in the control room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits every 12 hours in order to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs 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. Verification of boron concentration by performing a calculation based on level increase, RCS boron concentration, and last sample results; or by sampling the affected SIT within 6 hours, whenever a SIT is drained to maintain contained borated water level 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 is from the RWT, because the water contained in the RWT is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 5).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator ensures that an active failure could not result in the undetected closure of a SIT motor operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

SR 3.5.2.5 allows power to be supplied to the motor operated isolation valves when RCS pressure is < 1500 psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. UFSAR, Section 6.
3. 10 CFR 50.46.
4. UFSAR, Chapter 15.
5. NUREG-1366, December 1992.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 SITs – Shutdown

BASES

The functions of the four SITs 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.

The refill phase of a LOCA follows immediately where 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 the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer 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.

The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The SITs are passive components, since no operator or control action is required for them to perform their function. Internal tank pressure is sufficient to discharge the contents to the RCS, if RCS pressure decreases below the SIT pressure.

(continued)

BACKGROUND (continued)

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) Systems. Each SIT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

Additionally, the SIT motor operated isolation valves are interlocked with the pressurizer pressure instrumentation channels to ensure that the valves will automatically open as RCS pressure increases above SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow one less than the required SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that one less than the required SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

Due to the reduced decay heat removal requirements in MODES 3 and 4, and the reduced probability of a Design Basis Accident (DBA), the SITS operational requirements are reduced. The operational requirement allows either three or four SITs to be OPERABLE with a reduced borated water volume.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves and SIT nitrogen vent valves, however, are not single failure proof; therefore, whenever the SIT motor operated isolation valves are open, power is removed from their operators, and the switch is key locked open. Whenever the SIT vent valves are closed, power is removed with a keylock switch.

These precautions ensure that the SITs are available during an accident (Ref. 3). With power supplied to the valves, a single active failure could result in a valve failure, which would render one of the required SITs unavailable for injection. If a second required SIT is lost through the break, only the remaining required SIT(s) would reach the core. Active failures that could affect the SITs would be the closure of a motor operated outlet valve or opening of a solenoid operated nitrogen vent valve, the requirement to remove power from these eliminates this failure mode. Power is removed from the SIT isolation valves and nitrogen vent valves when pressurizer pressure is ≥ 1500 psia. This is consistent with the minimum LOCA analysis pressure of 1600 psia.

The minimum volume requirement for the required SITs, assuming one SIT is not available, ensures that the SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum amount of boron inventory in the SITs.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

For three OPERABLE SITs, the safety analysis uses a minimum of 1361 cubic feet of borated water and a maximum of 2000 cubic feet of borated water. To allow for instrument inaccuracy, a 60% wide range level (corresponding to 1451.5 cubic feet) and a 83% wide range level (corresponding to 1914 cubic feet) are specified. For four OPERABLE SITs, the safety analysis uses a minimum of 908 cubic feet of borated water and a maximum of 2000 cubic feet of borated water. To allow for instrument inaccuracy, a 39% wide range level (corresponding to 1029.2 cubic feet) and a 83% wide range level (corresponding to 1914 cubic feet) are specified. The percentage figures are provided in the LCO for operator use because the level indicator provided in the control room is marked in percentage, not in cubic feet.

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of 235 psig and a maximum pressure of 637 psig are used in the analyses. To allow for instrument accuracy, a 260 psig minimum and 625 psig maximum are specified. The maximum allowable boron concentration of 4400 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The 2300 ppm minimum boron concentration in the SITs assures that the back leakage from the RCS will not dilute the SITs below the minimum boron concentration in the safety analysis. The minimum safety analysis boron requirements of 2000 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the Refueling Water Tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS.

SIT-Shutdown satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODES 3 and 4 with pressurizer pressure less than 1837 psia, the LCO establishes the minimum conditions required to ensure that the required SITs are available to accomplish their core cooling safety function following a LOCA. The number of SITs required to be OPERABLE is based on the minimum required volume that will reach the core during a LOCA, assuming a single failure.

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of less than the remaining required tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For a required SIT to be considered OPERABLE, the motor operated isolation valve must be fully open when pressurizer pressure is ≥ 430 psia, power removed when pressurizer pressure is ≥ 1500 psia, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

(continued)

APPLICABILITY

In MODES 1 and 2, and MODES 3 and 4 with pressurizer pressure ≥ 1837 psia, the OPERABILITY requirements for SITs are covered by LCO 3.5.1.

In MODES 3 and 4 with pressurizer pressure < 1837 psia, the reduced borated water volume requirement is acceptable, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODE 4 with pressurizer pressure < 430 psia, the SIT motor operated isolation valves may be closed to isolate the SITs from the RCS but must remain energized. This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs. In this situation manual actions would be required to open the SIT motor operated isolation valves (i.e., manually initiated SIAS).

In MODES 5 and 6 the SITs are not required and the SIT motor operated isolation valves are closed as required to isolate the SITs from the RCS.

ACTIONS

A.1

If the boron concentration of one of the required SITs 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, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of the required SITs assuming a single failure, the consequences are less severe than they would be if a SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

(continued)

ACTIONS

B.1

If one SIT is inoperable for a reason other than boron concentration, the SIT must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three SITs 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 1 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 SIT to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1

If the inoperability of the required SIT was discovered but not restored while the plant was within the applicability of specification 3.5.1, "SITs - Operating", the plant must be brought to a MODE in which the LCO does not apply. The time allowed for restoration in specification 3.5.2 is adequate and may not be duplicated, for the same condition, when in specification 3.5.2, "SITs - Shutdown".

If the required SIT cannot be restored 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 5 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

D.1

If more than one of the required SITs is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification every 12 hours that each required SIT isolation valve is fully open when pressurizer pressure is ≥ 430 psia as indicated in the control room, ensures that the required SITs are available for injection and ensures timely discovery if a valve should be partially closed. If a required isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.2.2 and SR 3.5.2.3

Borated water volume and nitrogen cover pressure for the required SITs should be verified to be within specified limits every 12 hours in order to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.2.4

Thirty-one days is reasonable for verification to determine that each required SIT's boron concentration is within the required limits, because the static design of the SITs 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. Verification of boron concentration by performing a calculation based on level increase, RCS boron concentration, and last sample results; or sampling the affected SIT within 6 hours whenever a SIT is drained to maintain contained borated water level 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 is from the RWT,

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued)

because the water contained in the RWT is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4).

SR 3.5.2.5

Verification every 31 days that power is removed from each required SIT isolation valve operator when the pressurizer pressure is ≥ 1500 psia ensures that an active failure could not result in the undetected closure of a SIT motor operated isolation valve. If this were to occur, two less than the required SITs would be available for injection, given a single failure coincident with a LOCA.

Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 1500 psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. NUREG-1366, December 1992.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 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);

b. Control Element Assembly (CEA) 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.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs. After the blowdown stage of the LOCA stabilizes, injection flow is split equally between the hot and cold legs. After the Refueling Water Tank (RWT) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump.

Two redundant, 100% capacity trains are provided. In MODES 1, 2, and 3, with pressurizer pressure ≥ 1837 psia or with RCS T_c ≥ 485°F each train consists of High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) subsystems. In MODES 1, 2, and 3, with pressurizer pressure ≥ 1837 psia or with RCS Tc ≥ 485°F both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure.

(continued)

BASES

BACKGROUND (continued)

A suction header supplies water from the RWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI pump divide into four supply lines. Both HPSI trains feed into each of the four injection lines. The discharge header from each LPSI pump divides into two supply lines, each feeding the injection line to two RCS cold legs. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

The Safety Injection (SI) systems are actuated upon receipt of an SIAS. 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 Diesel Generators (DGs). 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.

The active ECCS components, along with the passive Safety Injection Tanks (SITs) and the RWT, covered in LCO 3.5.1, "Safety Injection Tanks (SITs)-Operating"; LCO 3.5.2, "SITs-Shutdown"; and LCO 3.5.5, "Refueling Water Tank (RWT)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 2) for ECCSs, 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;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 a Steam Line Break (SLB) and ensures that containment temperature limits are met.

Both HPSI and LPSI subsystems are assumed to be OPERABLE in the large break LOCA analysis at full power (Ref. 3). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pumps are credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The SGTR and SLB analyses also credit the HPSI pumps, but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. 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 CEA insertion during 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.

On smaller breaks, RCS pressure will stabilize at a value dependent upon break size, heat load, and injection flow. The smaller the break, the higher this equilibrium pressure. In all LOCA analyses, injection flow is not credited until RCS pressure drops below the shutoff head of the HPSI pumps.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the HPSI pump will deliver sufficient water during a small break LOCA and provide sufficient boron to maintain the core subcritical following an SLB.

ECCS – Operating satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODES 1, 2, and 3, with pressurizer pressure ≥ 1837 psia or with RCS T_c ≥ 485°F two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming there is a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ 1837 psia or with RCS T_c ≥ 485°F an ECCS train consists of a HPSI subsystem and a LPSI subsystem.

Each train includes the piping, instruments, valves, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the RWT on a SIAS and automatically transferring suction to the containment sump upon a Recirculation Actuation Signal (RAS).

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the RWT to the RCS, via the HPSI and LPSI pumps and their respective supply headers, to each of the four cold leg injection nozzles. In the long term (post RAS), this flow path is manually switched two to three hours after a LOCA to supply part of its HPSI flow to the RCS hot legs via the HPSI hot leg injection valves which connect to the Shutdown Cooling (SDC) suction nozzles.

(continued)

BASES

LCO (continued)

Simultaneous hot and cold leg injection will maintain core cooling and boric acid flushing following a large break LOCA.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure ≥ 1837 psia or with RCS T_c ≥ 485°F the ECCS OPERABILITY requirements for the limiting Design Basis Accident (DBA) 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 HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on plant power. The minimum T_c is based on the ECCS OPERABILITY requirements for a MODE 3 steam line break with a stuck rod and a single HPSI failure to prevent a return to power. The requirements of MODES 2 and 3, with RCS pressure ≥ 1837 psia or with RCS T_c ≥ 485°F, are bounded by the MODE 1 analysis.

The ECCS functional requirements of MODE 3, with RCS pressure < 1837 psia and with RCS T_c < 485°F, and MODE 4 are described in LCO 3.5.4, "ECCS – Shutdown."

In MODES 5 and 6, unit 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 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level."

(continued)

BASES

ACTIONS

A-1

With one LPSI subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPSI subsystem could result in loss of ECCS function. The Completion Time is reasonable to perform corrective maintenance on the inoperable LPSI subsystem.

B.1

If one or more trains are inoperable, except for reasons other than Condition A (one LPSI inoperable) and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC study (Ref. 4) using a reliability evaluation and is a reasonable amount of time to effect many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of 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. The intent of this Condition is to maintain a combination of OPERABLE equipment such that 100% of the ECCS flow equivalent to 100% of a single OPERABLE train remains available. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

(continued)

BASES

ACTIONS

B.1 (continued)

An event accompanied by a loss of offsite power and the failure of an emergency DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

C.1, C.2, and C.2

If the inoperable train cannot be restored 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 pressurizer pressure reduced to < 1837 psia and RCS Tc reduced to < 485°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

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 valves 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 automatically repositions 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 procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.2

With the exception of systems in operation, 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. One method of ensuring that any voids or pockets of gases are removed from the ECCS piping is to vent the accessible discharge piping high points, which is controlled by PVNGS procedures. 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. 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 SIAS or during SDC. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of 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 unit safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. The frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.5.3.4, SR 3.5.3.5, and SR 3.5.3.6

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. 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 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 the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

The following valve actuations must be verified at least once per 18 months:

- on an actual or simulated recirculation actuation signal, the containment sump isolation valves open, and the HPSI, LPSI and CS minimum bypass recirculation flow line isolation valves and combined SI mini flow valve close.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.7

Realignment of valves in the flow path on an SIAS is necessary for proper ECCS performance. The safety injection valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on current industry practice. These valves are also monitored in accordance with the requirements of 10 CFR 50.65 (Ref. 6).

SR 3.5.3.8

Periodic inspection of the containment sump ensures 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 an outage, on the need to have access to the location, and on the potential for unplanned transients if the Surveillance were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Chapter 6.
4. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
5. 10 CFR 50.65.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases B 3.5.3, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 3 with pressurizer pressure < 1837 psia and RCS T_c < 485°F, and in MODE 4, an ECCS train is defined as one High Pressure Safety Injection (HPSI) subsystem. The HPSI flow path consists of piping, valves, and pumps that enable water from the Refueling Water Tank (RWT) on a SIAS signal to be injected into the Reactor Coolant System (RCS) and automatically transferring HPSI suction to the containment sump on a Recirculation Actuation Signal (RAS) following the accidents described in Bases 3.5.3.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.3 is applicable to these Bases.

Due to the stable conditions associated with operation in MODE 3 with pressurizer pressure <1837 psia and with RCS T_c < 485°F and in MODE 4, and the reduced probability of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. Protection against single failures is not relied on for this MODE of operation.

ECCS – Shutdown satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

In MODE 3 with pressurizer pressure < 1837 psia and with RCS T_c < 485°F and in MODE 4 an ECCS subsystem is composed of a single HPSI subsystem. Each HPSI subsystem includes the piping, instruments, valves, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to supply water from the RWT to the RCS via the HPSI pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term post (RAS), this flow path is manually switched 2 to 3 hours after a LOCA to supply part of its HPSI flow to the RCS hot legs via the HPSI hot leg injection valves which connect to the Shutdown Cooling (SDC) suction nozzles.

With RCS pressure < 1837 psia and with RCS T_c < 485°F, one HPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. The Low Pressure Safety Injection (LPSI) pumps may therefore be released from the ECCS train for use in SDC.

APPLICABILITY

In MODES 1, 2, and 3 with RCS pressure ≥ 1837 psia or with RCS T_c ≥ 485°F, the OPERABILITY requirements for ECCS are covered by LCO 3.5.3.

In MODE 3 with RCS pressure < 1837 psia and with RCS T_c < 485°F and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit 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 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level."

(continued)

BASES

ACTIONS

A.1

With no HPSI pump OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The 1 hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity or to initiate actions to place the unit in MODE 5, where an ECCS train is not required.

B.1

When the Required Action cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The applicable Surveillance descriptions from Bases 3.5.3 apply as they pertain to the required HPSI train.

REFERENCES

The applicable references from Bases 3.5.3 apply as they pertain to the required HPSI train.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Refueling Water Tank (RWT)

BASES

BACKGROUND

The RWT supports the ECCS and the Containment Spray System by providing a source of borated water for Engineered Safety Feature (ESF) pump operation.

The RWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. A motor operated isolation valve is provided in each header to allow the operator to isolate the usable volume of the RWT from the ECCS after the ESF pump suction has been transferred to the containment sump following depletion of the RWT during a Loss of Coolant Accident (LOCA). A separate header is used to supply the Chemical and Volume Control System (CVCS) from the RWT. Use of a single RWT to supply both trains of the ECCS is acceptable since the RWT is a passive component, and passive failures are not assumed to occur coincidently with the Design Basis Event during the injection phase of an accident. Not all the water stored in the RWT is available for injection following a LOCA; the location of the ECCS suction piping in the RWT will result in some portion of the stored volume being unavailable.

The High Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the RWT, which vents to the Fuel Building Ventilation System. When the suction for the HPSI and containment spray pumps is transferred to the containment sump, this flow path must be isolated to prevent a release of the containment sump contents to the RWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

This LCO ensures that:

a. The RWT contains sufficient borated water to support the ECCS during the injection phase;

(continued)

BASES

BACKGROUND (continued)

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS 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 a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

The RWT also provides a source of borated water to the charging system for makeup to the RCS to compensate for contraction of the RCS coolant during plant cooldown while maintaining adequate shutdown margin. Although this charging system boration function is not required to be in a Technical Specification LCO per 10 CFR 50.36(c)(2)(ii) criteria, the RWT volume requirements of Figure 3.5.5-1 include this function in order to provide the plant operators with a single requirement for RWT volume.

For hot zero power temperature of 565 degrees F, the RWT volume requirement of 600,000 gallons will ensure adequate shutdown margin during a subsequent cooldown. For power levels greater than zero, with a corresponding increase in average RCS temperature, the volume of borated water to maintain the shutdown margin is the same as at zero power. Contraction requirements are greater at higher average RCS temperatures; however, the additional contraction is accommodated by an acceptable reduction in pressurizer level. Consequently, for operation at average RCS temperatures greater than 565 degrees F, the minimum volume required in the in the RWT is constant at 600,000 gallons.

(continued)

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI and containment spray 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 Bases B 3.5.3, "ECCS–Operating," and B 3.6.6, "Containment Spray." These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of Figure 3.5.5-1 for the ESF function is based on two factors:

a. Sufficient deliverable volume must be available to provide at least 20 minutes (plus a 10% margin) of full flow from all ESF pumps prior to reaching a low level switchover to the containment sump for recirculation; and

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switchover to recirculation occurs. This sump volume water inventory is supplied by the RWT borated water inventory.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Twenty minutes is the point at which 75% of the design flow of one HPSI pump is capable of meeting or exceeding the decay heat boiloff rate.

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate Net Positive Suction Head (NPSH) for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this amount of water without considering the inventory added from the safety injection tanks or Reactor Coolant System (RCS), but accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and to areas outside containment due to leakage from ECCS injection and recirculation equipment.

The 4000 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the Control Element Assembly (CEA) of highest worth, which is withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of core life.

The maximum boron limit of 4400 ppm in the RWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWT in excess of the limit could result in precipitation earlier than assumed in the analysis.

The upper limit of 120°F and the lower limit of 60°F on RWT temperature are the limits assumed in the accident

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

analysis. Although RWT temperature affects the outcome of several analyses, the upper and lower limits established by the LCO are not limited by any of these analyses.

The RWT ESF function satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The RWT 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) and to cool and cover the core in the event of a LOCA, that the reactor remains subcritical following a DBA, and that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the RWT must meet the limits established in the SRs for water volume, boron concentration, and temperature.

APPLICABILITY

In MODES 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the 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 RWT must 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 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With RWT boron concentration or borated water temperature not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of

(continued)

BASES

ACTIONS

A.1 (continued)

8 hours to restore the RWT to within limits was developed considering the time required to change boron concentration or temperature and that the contents of the tank are still available for injection and core cooling.

B.1

With RWT borated water volume not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The allowed Completion Time of 1 hour to restore the RWT to OPERABLE status is based on this condition since the contents of the tank are not available for injection and core cooling.

C.1 and C.2

If the RWT cannot be restored 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 3.5.5.1

RWT borated water temperature shall be verified every 24 hours to be within the limits assumed in the accident analysis. This Frequency has been shown to be sufficient to identify temperature changes that approach either acceptable limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1 (continued)

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT. With ambient temperatures within this range, the RWT temperature should not exceed the limits.

SR 3.5.5.2

The RWT water volume level shall be verified every 7 days in accordance with Figure 3.5.5-1. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation. Since the RWT volume is normally stable and is provided with a Low Level Alarm in the Control Room, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.5.3

Boron concentration of the RWT shall be verified every 7 days to be within the required range. This Frequency ensures that the reactor will remain subcritical following a LOCA and the boron precipitation in the core will not occur earlier than predicted. Further, it ensures that the resulting sump pH will be maintained in an acceptable range such that the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWT volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown through operating experience to be acceptable.

REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.6 Trisodium Phosphate (TSP)

BASES

BACKGROUND

Anhydrous Trisodium Phosphate (TSP) is placed on the floor of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a Loss of Coolant Accident (LOCA), remains in solution. TSP also helps inhibit Stress Corrosion Cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The emergency core cooling water is borated for reactivity control. This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and Containment Spray Systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

(continued)

BASES

BACKGROUND (continued)

Adjusting the Ph of the recirculation solution to levels at or above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH at or above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

Granular anhydrous TSP is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are placed on the floor of the containment building to dissolve from released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

LCO

The TSP is required to adjust the pH of the recirculation water to ≥ 7.0 after a LOCA. A pH ≥ 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH ≥ 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

(continued)

BASES

LCO (continued)

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of $\geq$ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the design basis value for density of anhydrous TSP. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

APPLICABILITY

In MODES 1, 2, and 3, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODES 4, 5, and 6, the potential for a LOCA is reduced and TSP is not required.

ACTIONS

A.1

If it is discovered that the TSP in the containment building is not within limits, action must be taken to restore the TSP to within limits.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of 18 months is required to determine visually that a minimum of 524 cubic feet is contained in the TSP baskets (Ref. 1). This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0.

The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.5.6.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of 3.5 grams ± 0.005 grams of TSP from one of the baskets in containment is submerged in 1.0 liters ± 0.005 liters of 2.5 wt% boric acid solution (nominally at 4400 ppm) at 135°F ± 9°F. Without agitation, the solution pH as measured at 77°F ± 9°F should be raised to ≥ 7 within 4 hours. The representative sample weight is based on the minimum required TSP weight of 25,325 pounds which at installed density corresponds to the minimum volume of 524 cubic ft,

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.6.2 (continued)

and maximum possible post LOCA sump volume of 867,124 gallons, normalized to buffer a 1.0 liter sample.

The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post LOCA sump volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH $\geq$ 7.0 by the onset of recirculation after a LOCA.

The temperature of 135 $\pm$ 9°F was chosen for the borated water solution because that is the minimum temperature at the inlet of the shutdown cooling heat exchangers, therefore, this would be the lowest temperature possible during this type of accident.

REFERENCES

1. PVNGS operating license amendment numbers 110, 102 and 82 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated December 10, 1996.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete Containment Building (CB), its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a two way pattern of tendons, which are an extension of the continuous vertical tendons. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete CB is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

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

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

(continued)

BASES (continued)

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of 52.0 psig, which results from the limiting design basis LOCA.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

(continued)

BASES (continued)

LCO (continued)

Type A leakage rate testing measures the overall leakage rate of the containment. Type B leakage rate testing measures the local leakage rate of blind flanges, air locks and other devices which employ resilient seals. Type C leakage rate testing measures the local leakage rate of valves. Refer to reference 1 for a more detailed definition.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and purge valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required 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 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 3.8.
3. UFSAR, Section 6.2.
4. Regulatory Guide 1.35, Revision 1 and Draft of Revision 3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 9 ft.-6 inches in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide local indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a [52 psig], following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. If the inner door is inoperable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable because of the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions. A third Note has been included that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage limit.

(continued)

BASES

ACTIONS (continued)

A.1, A.2, and A.3

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour. Action A applies to any condition which affects only one side of the air lock such that closure of the opposite door maintains containment OPERABILITY. Examples of an inoperable air lock door are cracked viewglass, equalizing valve leaking, or door seals leaking.

In addition, the affected air lock penetration must be isolated by locking closed an OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

(continued)

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

(continued)

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required 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 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program and includes testing of the airlock doors following each closing, as specified.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type Band C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when containment is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 3.8.
3. UFSAR, Section 6.2.
4. UFSAR, Section 15.6

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

Containment isolation occurs upon receipt of a high containment pressure signal or a low pressurizer pressure signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of Engineered Safety Feature Systems in order to prevent leakage of radioactive material. Upon actuation of safety injection, automatic containment isolation valves also isolate systems not required for containment or RCS heat removal. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the accident analysis will be maintained.

(continued)

BASES

BACKGROUND (continued)

The purge valves were designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor RCS leakage prior to personnel entry into containment. There are two sets of purge valves: refueling purge valves and power access purge valves. The refueling and power access supply and exhaust lines are each supplied with inside and outside containment isolation valves but share common supply and exhaust headers.

The refueling purge valves are designed for purging the containment atmosphere to the unit stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the unit is shut down during refueling operations and maintenance. Motor operated isolation valves are provided inside and outside the containment. The valves are operated manually from the control room. The valves will close automatically upon receipt of a containment purge isolation actuation signal and a containment isolation actuation signal. Because of their large size, the refueling purge valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, the refueling purge valves are maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Open refueling purge valves, or a failure of the power access purge valves to close, following an accident that releases contamination to the containment atmosphere would cause a significant increase in the containment leakage rate.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly ejection accident. In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the refueling purge valves are closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources.

The refueling purge valves may be unable to close in the environment following a LOCA. Therefore, each of the refueling purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment refueling purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO. The power access purge valves are capable of closing under accident conditions. Therefore, they are allowed to be open for limited periods during power operation.

The OPERABILITY of main steam safety valves, main steam isolation valves, main feedwater isolation valves, and main steam atmospheric dump valves is covered by Specifications 3.7.1, 3.7.2, 3.7.3 and 3.7.4 respectively.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The refueling purge valves must be maintained sealed closed. The valves covered by this LCO are listed with their associated stroke times in the UFSAR (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference 2. ESF actuated valves are considered OPERABLE when locked, sealed, or otherwise prevented from unintended operation, in their actuated position.

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service afer maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

Opening of vent, drain, and test connections located between the inboard and outboard Containment Isolation Valve (CIV) could result in a loss of OPERABILITY for the affected penetration flow path. The appropriate actions for the associated CIV must be met for these connections.

The ACTIONS are modified by a Note allowing penetration flow paths, except for 42 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment refueling purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

A fourth Note has been added that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable except for purge valve leakage not within limit (refer to Action D), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable except for purge valve leakage not within limit (refer to Action D), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

(continued)

BASES

ACTIONS

B.1 (continued)

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths which are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere (10 CFR 150, APP. A, GDC 57).

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1, D.2, and D.3

In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve with resilient seals, or a blind flange. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment capable of being mispositioned are in the correct position.

(continued)

BASES

ACTIONS

D.1, D.2 and D.3 (continued)

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with a resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 3). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 42 inch containment purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the refueling purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 4), related to containment purge valve use during unit operations. This SR is not required to be met while in Condition D of this LCO. This is reasonable since the penetration flow path would be isolated.

SR 3.6.3.2

This SR ensures that the power access purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The power access purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, 4 and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 5), is required to ensure OPERABILITY. Industry operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.6 (continued)

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on an actual or simulated actuation 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 was developed considering it is prudent that this SR be performed only during a unit outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.2.4.
2. UFSAR, Section 6.2.6.
3. Generic Issue B-20.
4. Generic Issue B-24.
5. 10 CFR 50, Appendix J, Option B.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. A double ended discharge line break LOCA with maximum ECCS results in the highest calculated internal containment pressure of 52.0 psig, which is below the internal design pressure of 60 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) Systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis bounds the containment pressure allowed during normal operation. The LCO limit of 2.5 psig ensures that, in the event of an accident, the maximum peak containment internal pressure, 52.0 psig, and the maximum accident design pressure for containment, 60 psig, are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an internal pressure equal to 4.0 psig below external pressure in order to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of -0.3 psig ensures that operation within the design limit of -4.0 psig is maintained. The maximum calculated external pressure that would occur as a result of an inadvertent actuation of the Containment Spray System is 2.6 psig.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative pressure differential following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

(continued)

BASES

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required 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 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the accident analysis. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray System during post accident conditions is dependent on the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis (Ref. 1). Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst case LOCA generates larger mass and energy releases than the worst case MSLB; however, the MSLB event results in a higher peak temperature than the LOCA event. The initial pre-accident temperature inside containment was assumed to be 120°F (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The initial containment average air temperature condition of 120°F resulted in a maximum vapor temperature in containment of 398.5°F. The temperature of the containment steel liner and concrete structure reach approximately 230°F and 220°F, respectively. The containment average air temperature limit of 120°F ensures that, in the event of an accident, the maximum design temperature for containment, 300°F, is not exceeded. The consequence of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.

The LCO limit of 117°F has been derived to account for instrument inaccuracies. The indicated limit of 117°F ensures that the actual limit of 120° will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required 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 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1 (continued)

The Primary containment average air temperature is determined by taking the arithmetical average of the temperatures at any five of the following locations:

a. Nominal Elevation 85'- 0"
b. Nominal Elevation 85'- 0"
c. Nominal Elevation 126'- 0"
d. Nominal Elevation 126'- 0"
e. Nominal Elevation 145' - 0"
f. Nominal Elevation 188' - 0"
g. Nominal Elevation 188' - 0"

REFERENCES

1. UFSAR, Section 6.2
2. UFSAR, Section 9.4

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray System is designed to the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Spray System is an Engineered Safety Feature (ESF) System. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, a shutdown cooling heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The Refueling Water Tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT to the containment sump(s).

The Containment Spray System provides a spray of cold borated water into the upper regions of containment to reduce containment pressure and temperature, to provide hydrogen mixing, and to reduce the concentration of fission products in the containment atmosphere during a DBA. The RWT solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In both the

(continued)

BASES

BACKGROUND (continued)

injection phase and the recirculation mode of operation, heat is removed from the spray water by the shutdown cooling heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 100% of the system design requirements for containment heat removal and 100% of the iodine removal design bases.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. An automatic actuation starts the two Containment Spray System pumps, opens the containment spray header isolation valves and begins the injection phase. A manual actuation of the Containment Spray System is available on the main control board to begin the same sequence. The injection phase continues until an RWT level Low signal is received. The Low level for the RWT generates a recirculation actuation signal that aligns valves from the containment spray pump suction to the containment sump. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Hydrogen mixing within the containment is accomplished by the Containment Spray System and the containment internal structure design, which permits convective mixing and prevents entrapment. The Containment Spray System prevents localized accumulations of hydrogen.

The Containment Spray System reduces the potential for breach of containment due to a hydrogen oxygen reaction by providing a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a local pocket of hydrogen above the flammable concentration and giving the operator the capability of preventing the occurrence of a bulk hydrogen burn inside containment per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light- Water-Cooled Reactors" (Ref. 7), and 10 CFR 50, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The Containment Spray System accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

APPLICABLE SAFETY ANALYSES

The Containment Spray System limits the temperature and pressure that could be experienced following a DBA. The Containment Spray System is required to be capable of reducing containment pressure to 1/2 the peak pressure within 24 hours following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the Loss Of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 52.0 psig (experienced during a LOCA). The analysis shows that the peak containment vapor temperature is 398.5°F (experienced during a MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 102% RTP, one containment spray train operating, and initial (pre-accident) conditions of 120°F and 16.7 psia. The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to -2.6 psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specification 3.6.4.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieve full flow through the containment spray nozzles. The Containment Spray System total response time of 91 seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

The Containment Spray System mixes the containment atmosphere to provide a uniform hydrogen concentration. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solution.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 8 are used to maximize the amount of hydrogen calculated.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

During a DBA, one containment spray train is required to maintain the containment peak pressure and temperature below the design limits (Ref. 5), to remove iodine from the containment atmosphere to maintain concentrations below those assumed in the safety analysis, and provide hydrogen mixing. To ensure that these requirements are met, two containment spray trains must be OPERABLE.

(continued)

BASES

LCO (continued)

Therefore, in the event of an accident, the minimum requirements are met, assuming that the worst case single active failure occurs.

Each Containment Spray System typically includes a spray pump, a shutdown cooling heat exchanger, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT upon an ESF actuation signal and automatically transferring suction to the containment sump.

APPLICABILITY

In MODES 1, 2, and 3, and Mode 4 with RCS pressure ≥ 385 psia, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment spray trains.

In MODE 4 with RCS pressure < 385 psia and MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in these MODES.

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray train is adequate to perform the iodine removal, hydrogen mixing, and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required 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 4 with RCS pressure < 385 psia within 84 hours. The allowed 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 plant systems. The extended interval to reach MODE 4 with RCS pressure < 385 psia allows additional time for the restoration of the containment spray train and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With two containment spray trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation (positioned to take suction from the RWT on a containment spray actuation test signal [CSAS]). 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 being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verifying, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Verifying that the containment spray header piping is full of water to the 113 ft level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The analyses shows that the header may be filled with unborated water which helps to reduce boron plate out due to evaporation. The 31 day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of water level in the piping occurring between surveillances. The value of 113' is an indicated value which accounts for instrument uncertainty.

SR 3.6.6.3

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow (either full flow or miniflow as conditions permit). This test is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 and SR 3.6.6.5 (continued)

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated safety injection actuation signal, recirculation actuation signal and containment spray actuation signal as applicable. 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 an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.5.3.5. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.6

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. UFSAR, Section 6.5.
4. UFSAR, Section 7.3.
5. UFSAR, Section 3.1.34
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. 10 CFR 50.44.
8. Regulatory Guide 1.7, Revision 0.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction. Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and 10 CFR 50, GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiners are shared among the three units. Each consists of controls, a power supply, and a recombiner located in the Auxiliary Building. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at 50 cfm with a 5 hp centrifugal blower in the unit providing the motive force. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analysis are not exceeded and minimizing damage to safety

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

related equipment located in containment. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended in Reference 3 are used to maximize the amount of hydrogen calculated.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two hydrogen recombiners shared among the three units must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the post LOCA hydrogen concentration within containment below its flammability limit of 4.0 v/o, assuming a worst case single failure.

(continued)

BASES

APPLICABILITY (continued)

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

The required ACTIONS have been modified by a Note stating that all three PVNGS Units (Units 1, 2, and 3) shall simultaneously comply with the REQUIRED ACTION(s) when the shared portion of the hydrogen recombiner(s) is the cause of a CONDITION. This is necessary since the three PVNGS Units share the two hydrogen recombiners that are required by this LCO. It will be necessary for the Control Room of the Palo Verde Unit that discovers an inoperable shared portion of the hydrogen recombiner(s) to notify the other two Palo Verde Unit's Control Rooms of the inoperability.

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or MSLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or MSLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or MSLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or MSLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the Hydrogen Purge Cleanup System. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required 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. The allowed 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 plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

This SR ensures that there are no physical problems that could affect recombiner operation. A visual inspection is sufficient to determine abnormal conditions that could cause failures. The 6 month Frequency for this SR was developed considering that the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.2

A functional test of each Hydrogen Recombiner System assures that the recombiners remain operational. The functional test shall include operating the recombiner including the air blast heat exchanger fan motor and enclosed blower motor continuously for at least 30 minutes at a temperature of approximately 800°F reaction chamber temperature. The frequency recommended for this surveillance in the Improved Standard Technical Specifications (NUREG-1432, Rev. 1) is 18 months. The bases for NUREG 1432 was developed for permanently installed hydrogen recombiners. The two portable hydrogen recombiners at PVNGS are shared among the three units; therefore, the 6 month frequency from the initial licensing basis is retained for reliability considerations.

SR 3.6.7.3

Performance of a CHANNEL CALIBRATION to include a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires 1) resistance checks of motors, thermocouples, and heater systems, 2) testing/calibration of all flow elements, switches, and temperature elements, and 3) operation of the recombiner to include a functional test at 1200°F (±50°F) for at least 4 hours. Operating experience has shown that these components usually pass the Surveillance when performed at the 12 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision 0.
4. UFSAR, Section 6.2.5

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Reactor Coolant Pressure Boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each of the four main steam lines, outside containment, upstream of the main steam isolation valves, as described in the CESSAR, Section 5.2 (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering if there is insufficient steam pressure to fully open all valves.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power Loss Of Condenser Vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater to the steam generators, RCS pressure reaches ≤ 2742 psia. This peak pressure is < 110% of the design pressure of 2500 psia, but high enough to actuate the pressurizer safety valves. The maximum relieving rate during the LOCV event is 14.5 E6 lb/hour, which is less than the rated capacity of 16 MSSVs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2843 psia, with the pressurizer safety valves providing relief capacity. These results were found acceptable by the NRC based on the low probability of the event. The maximum relieving rate of the MSSVs during the FWLB event is ≤ 3.8 E6 lb/hour, which is less than the rated capacity of four MSSVs.

The MSSVs satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.2, and Required Action A.3 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

(continued)

BASES

APPLICABILITY

In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

$$\text{Allowable THERMAL POWER} = \frac{(10 - N)}{10} \times 109.2$$

With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.

SP = Allowable THERMAL POWER + 9.8

where:

SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = Total number of MSSVs per steam generator.

N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.

109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).

9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than six MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ASME OMc (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a $\pm$ 3% setpoint tolerance for OPERABILITY; however, the valves are reset to $\pm$ 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. CESSAR, Section 5.2.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. UFSAR, Section 15.2.

4. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-3500.

5. ASME OMc, 1994 Addenda to the ASME OM Code 1990.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a High Energy Line Break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the Main Steam Safety Valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass Control System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either low steam generator pressure, high steam generator level or high containment pressure. The MSIVs fail closed on loss of control or actuation power. The MSIS also actuates the Main Feedwater Isolation Valves (MFIVs) to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in the CESSAR, Section 6.2 (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the UFSAR, Section 15.1.5 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment analysis is the hot zero power SLB inside containment with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam line to close. At zero power, the steam generator inventory and temperature are at their maximum,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

maximizing the analyzed mass and energy release to the containment. Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers, which are downstream of the other MSIVs. With the most reactive control element assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of an MFIV to close, and failure of an emergency diesel generator to start.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.

With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed. Significant single failures considered include: failure of a MSIV to close, failure of an emergency diesel generator, and failure of a HPSI pump.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into the containment, the analysis assumes that the MSIV in the affected steam line remains open. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIVs in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator, and from the residual steam in the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

main steam header downstream of the closed MSIVs in the intact loops.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break, and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that the MSIV in each of the four steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. The MSIVs have redundant actuator trains. An MSIV is OPERABLE with one train of hydraulics unavailable to shut the valve. Only one OPERABLE MSIV is allowed to have an unavailable hydraulic train.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

(continued)

BASES (continued)

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2, 3 and 4 except when all MSIVs are closed when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1 and A.2

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The 4 hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 4 hour Completion Time is consistent with that normally allowed for containment isolation valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If the MSIV cannot be restored to OPERABLE within 4 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 4 hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

(continued)

BASES (continued)

ACTIONS

D.1 and D.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is ≤ 4.6 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be full stroke tested at power.

The Frequency for this SR is in accordance with the Inservice Testing Program. This Frequency demonstrates the valve closure time at least once per refueling cycle.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. UFSAR, Section 10.3.
2. CESSAR, Section 6.2.
3. UFSAR, Section 15.1.5.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The MFIVs isolate Main Feedwater (MFW) flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFIVs terminates flow to both steam generators, terminating the event for Feedwater Line Breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Steam Line Breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs isolate the nonsafety related portions from the safety related portion of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide an additional pressure boundary for the controlled addition of Auxiliary Feedwater (AFW) to the intact loop.

Two MFIVs are located on each economizer and downcomer line, outside, but close to, containment. The downcomer MFIVs are located upstream of the train A and B AFW injection points so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the downcomer MFIVs to the steam generator must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs close on receipt of a Main Steam Isolation Signal (MSIS) generated by either low steam generator pressure, high steam generator level, or high containment pressure. The MSIS also actuates the Main Steam Isolation Valves (MSIVs) to close. The MFIVs may also be actuated manually. In addition to the MFIVs, check valves are available to isolate the feedwater line penetrating containment, and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.A description of the MFIVs is found in the UFSAR, Section 10.4.7 (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs may also be relied on to terminate a steam break for core response analysis and an excess feedwater flow event upon receipt of a MSIS on high steam generator level.

Failure of an MFIV to close following an SLB, FWLB, or excess feedwater flow event can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the nonsafety related portions from the safety related portions of the system. This LCO requires that two MFIVs in each feedwater line be OPERABLE. The MFIVs are considered OPERABLE when the isolation times are within limits, and are closed on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

(continued)

BASES

APPLICABILITY

The MFIVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator.

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE, except when they are closed or isolated by a closed power operated valve, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed or isolated by a closed power operated valve, they are already performing their safety function.

In MODES 5 and 6, steam generator energy is low. Therefore, the MFIVs are not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each penetration flow path.

A.1 and A.2

With one MFIV inoperable, action must be taken to close or isolate the inoperable valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable MFIVs that are closed to comply with Required Action A.1 must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The seven day completion time is responsible, based on engineering judgement, MFIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

(continued)

BASES

ACTIONS

B.1 and B.2

If more than one MFIV in the same flow path cannot be restored to OPERABLE status, then there may be no system to operate automatically and perform the required safety function. Under these conditions, valves in each flow path must be restored to OPERABLE status, closed, or the flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable to close an MFIV or otherwise isolate the affected flow path.

Inoperable MFIVs that cannot be restored to OPERABLE status within the Completion Time, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated.

C.1 and C.2

If the MFIVs cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR ensures the actuation of each MFIV is ≤ 9.6 seconds on an actual or simulated actuation signal. The MFIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MFIVs should not be full stroke tested at power.

The Frequency is in accordance with the Inservice Testing Program. The Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the specified Frequency.

REFERENCES

1. UFSAR, Section 10.4.7.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a safety grade method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass Control System to the condenser not be available, as discussed in the FSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the Condensate Storage Tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown.

Four ADV lines are provided. Each ADV line consists of one ADV and an associated block valve. One ADV line per steam generator is required to meet the assumptions in the safety analyses. The ADV block valves are not required to be closed in the event of a stuck open ADV.

The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for RCS cooldown to the SDC System entry conditions.

A description of the ADVs is found in Reference 1. The ADVs require both DC sources and class AC instrument power to be considered OPERABLE. In addition, hand wheels are provided for local manual operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to SDC System entry conditions. A cooldown rate of 75°F per hour is obtainable by one or both steam generators. This design is adequate to cool the unit to SDC System entry conditions with only one ADV and one steam generator, utilizing the cooling water supply available in the CST.

In the accident analysis presented in the UFSAR, the ADVs are assumed to be used by the operator to cool down the unit to SDC System entry conditions for accidents accompanied by a loss of offsite power. Prior to the operator action, the Main Steam Safety Valves (MSSVs) are used to maintain steam generator pressure and temperature at the MSSV setpoint. This is typically 30 minutes following the initiation of an event. (This may be less for a Steam Generator Tube Rupture (SGTR) event.) The limiting events are those that render one steam generator unavailable for RCS heat removal, with a coincident loss of offsite power; this results from a turbine trip. Typical initiating events falling into this category are a main steam line break upstream of the main steam isolation valves, a feedwater line break, and an SGTR event (although the ADVs on the affected steam generator may still be available following a SGTR event).

The ADVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

One ADV line is required to be OPERABLE on each steam generator to conduct a unit cooldown following an event in which one steam generator becomes unavailable. Failure to meet the LCO can result in the inability to cool the unit to SDC System entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass Control System.

An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODES 5 and 6, an SGTR is not a credible event.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one required ADV line inoperable, action must be taken to restore the OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the availability of a nonsafety grade backup in the Steam Bypass Control System and MSSVs.

B.1

With two required ADV lines inoperable (one in each steam generator), action must be taken to restore one of the ADV lines to OPERABLE status. As the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass Control System and MSSVs, and the low probability of an event occurring during this period that requires the ADV lines.

(continued)

BASES

ACTIONS

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)") and pump to the steam generator secondary side via the main feedwater (MFW) piping. The discharge piping from the two essential AFW pumps is cross connected outside containment. The AFW lines then penetrate containment and connect to the downcomer piping. The non-essential AFW pump discharge piping splits with a line connecting with each downcomer line outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.

The AFW System consists of one essential motor driven AFW pump, one non-essential motor driven AFW pump, and one essential steam turbine driven pump configured into three trains. Each essential pump provides 100% of AFW flow capacity to the steam generators as assumed in the accident analysis. The non-essential pump is not capable of providing 100% capacity with the recirc line open. All three pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

The essential motor driven AFW pump is powered from an independent Class 1E power supply, and has the capability to be realigned from the control room to feed either steam generator. The non-essential motor driven AFW pump is powered from a Class 1E power supply and can be aligned to feed either steam generator. This pump is manually activated.

One essential pump provides sufficient flow to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

(continued)

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from either main steam header upstream of the main steam isolation valve (MSIV). Each of the steam feed lines is capable of supplying 100% of the requirements of the turbine driven AFW pump. The turbine driven AFW pump is capable of feeding either steam generator, with DC powered control valves actuated to the appropriate steam generator by the Auxiliary Feedwater Actuation Signal (AFAS).

The non-essential AFW train supplies feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

For the normal plant conditions stated above, the non-essential AFW train is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at no load conditions ($\leq$ 1170 psia). Subsequently, the non-essential AFW train supplies sufficient water to cool the unit to SDC entry conditions.

The AFW System actuates automatically on low steam generator level by the AFAS as described in LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The AFAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW System from a steam generator having a significantly lower steam pressure than the other steam generator. The AFAS automatically actuates the AFW turbine driven pump and associated DC operated valves and controls when required, to ensure an adequate feedwater supply to the steam generators. DC operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in the FSAR, Section 10.4.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the essential AFW trains is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to 1270 psia at the entrance to the steam generators.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB); and

b. Main Steam Line Break (MSLB).

c. Station Blackout

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valve and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, the AFAS logic might not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. The non-essential motor driven AFW pump, if started manually, would deliver to the broken down comer header at the pump runout flow until the problem was detected, and flow was terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the essential motor driven AFW pump.

The AFW System satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

This LCO requires that three AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Two essential and one non-essential AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering the essential motor driven AFW pump from an emergency bus. The non-essential motor driven AFW pump can be manually loaded on its emergency bus.

The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each capable of supplying AFW to either steam generator. The turbine driven AFW pump shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to either of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

Although the operability of the non-essential motor driven AFW pump is important from a risk persepctive, this pump is not credited in the PVNGS Accident Analyses.

The LCO is modified by a Note indicating that only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam supply available in MODE 4 to power the turbine driven AFW pump.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the MFW System is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4, the AFW System may be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW pumps is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event requiring the inoperable steam supply to the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS
(continued)

B.1

With one of the required AFW trains (pump or flow path) inoperable, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When either Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODES 1, 2, and 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4, with two AFW trains inoperable, operation is allowed to continue because only one motor driven AFW pump (either the essential or the non-essential pump) is required in accordance with the Note that modifies the LCO. Although it is not required, the unit may continue to cool down and start the SDC.

(continued)

BASES

ACTIONS
(continued)

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. Completion Times are also suspended at the time the conditions is entered. The Completion Time is resumed with the time remaining when the Condition was entered upon restoration of one AFW train to OPERABLE status.

With all three AFW trains inoperable in MODES 1, 2, and 3, the unit is in a seriously degraded condition with no TS related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. Completion Times are also suspended at the time the Condition is entered. The Completion Time is resumed with the time remaining when the Condition was entered upon restoration of one AFW train to OPERABLE status.

With one AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status or to immediately verify, by administrative means, the OPERABILITY of a second train. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

In MODE 4, either the reactor coolant pumps or the SDC loops can be used to provide forced circulation as discussed in LCO 3.4.6, "RCS Loops - MODE 4."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW water and steam supply flow paths provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Section XI of the ASME Code (Ref.2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing may be performed on recirculation flow. This test confirms one point on the pump design curve and can be indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME Code, Section XI (Ref. 2), at 3 month intervals satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. Normal operating pressure is established in the steam generators when RCS temperature reaches 532°F, this corresponds to a P_{sat} of 900 psia. This deferral is required because there is an insufficient steam pressure to perform the test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is not required for the non-essential train since there are no automatic valves which receive an AFAS. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. Normal operating pressure is established in the steam generators when RCS temperature reaches 532°F, this corresponds to a P_{sat} of 900 psia. This deferral is required because there is an insufficient steam pressure to perform the test.

Also, this SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.5.4

This SR ensures that the essential AFW pumps will start in the event of any accident or transient that generates an AFAS signal by demonstrating that each essential AFW pump starts automatically on an actual or simulated actuation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

The non-essential AFW pump does not automatically activate and is not subject to this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4 (continued)

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. Normal operating pressure is established in the steam generators when RCS temperature reaches 532°F, this corresponds to a P_{sat} of 900 psia. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump is already operating and the autostart function is not required.

SR 3.7.5.5

This SR ensures that the AFW System is properly aligned by verifying the flow path from each essential AFW pump to each steam generator prior to entering MODE 2 operation, after 30 days in MODE 5 or 6. OPERABILITY of essential AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and administrative controls to ensure that flow paths remain OPERABLE.

To further ensure AFW System alignment, the OPERABILITY of the essential AFW flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of 650 gpm at 1270 psia at the entrance to the steam generators. (This SR is not required for the non-essential AFW pump since it is normally used for startup and shutdown.)

REFERENCES

1. UFSAR, Section 10.4.9.

2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST is the primary source of water for the Auxiliary Feedwater (AFW) System (LCO 3.7.5, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) or the atmospheric dump valves.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass control valves. The condensed steam is returned to the CST by the condensate pump draw-off. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from the Reactor Makeup Water Tank (RMWT).

A description of the CST is found in the UFSAR, Section 9.2.6 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the CST has sufficient volume to maintain the plant for 8 hours at MODE 3, followed by a cooldown to shutdown cooling (SDC) entry conditions at the design cooldown rate.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting event for the condensate volume is the large feedwater line break with a coincident loss of offsite power. Single failures that also affect this event include the following:

a. The failure of the diesel generator powering the motor driven AFW pump (requiring additional steam to drive the remaining AFW pump turbine); and

b. The failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break either in the main feedwater, or essential AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, as the Auxiliary Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated by the retaining of steam generator inventory. A break in the non-essential AFW line could have similar consequences on CST level but is not controlled by AFAS. Actuation required operator action.

The CST satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 4 hours following a reactor trip from 102% RTP, and then cool down the RCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown, as well as to account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The CST level required is a usable volume of ≥ 300,000 gallons, which is based on holding the unit in MODE 3 for 8 hours, followed by a cooldown to SDC entry conditions at 75°F per hour. This basis exceeds the level required by the NRC Standard Review Plan Branch Technical Position, Reactor

(continued)

BASES

LCO (continued)

Systems Branch 5-1 (Ref. 4).

OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST level is not within the limit, the OPERABILITY of the backup water supply (RMWT) must be verified by administrative means within 4 hours.

OPERABILITY of the RMWT must include verification of the OPERABILITY of flow paths from the RMWT to the AFW pumps, and availability of 26 ft. (300,000 gal.) of water in the RMWT. The CST level must be returned to OPERABLE status within 7 days, as the RMWT may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the RMWT. The 7 day Completion Time is reasonable, based on an OPERABLE RMWT being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. (This level ≥ 29.5 ft (300,000 gallons)). The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST level deviations.

REFERENCES

1. UFSAR, Section 9.2.6.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. NRC Standard Review Plan Branch Technical Position (BTP) RSB 5-1.

B 3.7 PLANT SYSTEMS

B 3.7.7 Essential Cooling Water (EW) System

BASES

BACKGROUND

The EW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. The EW System acts as a backup to the non-safety related Nuclear Cooling Water System for several non-safety related loads. The EW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Spray Pond System (ESPS), and thus to the environment.

The EW System is arranged as two independent full capacity cooling loops, which are normally isolated from the Nuclear Cooling Water System. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, chemical addition tank, and instrumentation. Each safety related train is powered from a separate bus. The surge tank in the system provides pump trip protective functions to ensure sufficient net positive suction head is available. The pump in each train is automatically started on receipt of an ESFAS signal.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2, Reference 1, and Section 9.2.1, Reference 2. The principal safety related function of the EW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchanger.

APPLICABLE SAFETY ANALYSES

The design basis of the EW System is for one EW train in conjunction with the ultimate heat sink and a 100% capacity Containment Spray System to remove sufficient heat to ensure a safe reactor shutdown coincident with a loss of offsite power. The EW System provides a gradual reduction in the temperature of the containment sump fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The EW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The EW System also functions to cool the unit from SDC entry conditions (T_{cold} < 350°F) to MODE 5 (T_{cold} < 210°F) during normal and post accident operations. The time required to cool from 350°F to 210°F is a function of the number of EW and SDC trains operating. One EW train is sufficient to remove decay heat during subsequent operations with T_{cold} < 210°F. This assumes that the worst case meteorological conditions occur simultaneously with the maximum heat loads on the system.

The EW System satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The EW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one EW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two EW trains must be OPERABLE. At least one EW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A EW train is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of EW from other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EW System.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the EW System must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the EW System are determined by the systems it supports.

Reference LCO 3.0.6 and the SFDP (Specification 5.5.15).

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for SDC made inoperable by EW. This note is only applicable in Mode 4. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

With one EW train inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE EW train is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the EW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EW flow path provides assurance that the proper flow paths exist for EW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the EW components or systems may render those components inoperable but does not affect the OPERABILITY of the EW System.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the EW valves on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.3

This SR verifies proper automatic operation of the EW pumps on an actual or simulated actuation signal. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.2.

2. UFSAR, Section 9.2.1.

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Spray Pond System (ESPS)

BASES

BACKGROUND

The ESPS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During a normal shutdown, the ESPS also provides this function for various safety related components.

The ESPS consists of two separate, 100% capacity safety related cooling water trains. Each train consists of one 100% capacity pump, one Essential Cooling Water (EW) heat exchanger, piping, valves, instrumentation, and a cleanup and Chemistry Control System. The valves are manually aligned, and secured in position. The pumps are automatically started upon receipt of an ESFAS signal.

Additional information about the design and operation of the ESPS, along with a list of the components served, is presented in the FSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the ESPS is the removal of decay heat from the reactor via the EW System.

APPLICABLE SAFETY ANALYSES

The design basis of the ESPS is for one ESPS train, in conjunction with the EW System and a 100% capacity containment spray system to remove sufficient heat to ensure a safe reactor shutdown coincident with a loss of offsite power. The ESPS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ESPS, in conjunction with the EW System, also cools the unit from shutdown cooling (SDC), as discussed in the UFSAR, Section 5.4.7 (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of EW and SDC System trains that are operating. One ESPS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes that worst case meteorological conditions occur simultaneously with maximum heat loads on the system.

(continued)

BASES

APPLICABLE (continued)

The ESPS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two ESPS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power.

An ESPS train is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, instrumentation, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESPS System is required to support the OPERABILITY of the equipment serviced by the ESPS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESPS are determined by the systems it supports.

Reference LCO 3.0.6 and the SFDP (Specification 5.5.15).

ACTIONS

A.1

With one ESPS train inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ESPS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the ESPS train could result in loss of ESPS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1, "AC Sources—Operating," must be entered when the inoperable ESPS train results in an inoperable emergency diesel generator. The second Note

(continued)

BASES

ACTIONS

A.1 (continued)

indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops–MODE 4," should be entered if an inoperable ESPS train results in an inoperable SDC System. This note is only applicable in MODE 4. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the ESPS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated, valves in the ESPS flow path ensures that the proper flow paths exist for ESPS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the ESPS components or systems may render those components inoperable but does not necessarily affect the OPERABILITY of the ESPS. Although the ESPS remains operable, the effect of isolating components on the mechanical performance must be evaluated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

The SR verifies proper automatic operation of the ESPS pumps on an actual or simulated actuation signal. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.1.
2. UFSAR, Section 5.4.7.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. This is done utilizing the Essential Spray Pond System (ESPS).

The UHS is the essential spray ponds as discussed in the UFSAR, Section 9.2.5 (Ref. 1). The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that a 26 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on shutdown cooling. Its maximum post accident heat load occurs 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation, and the containment spray system is required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure. The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS. The 26 day supply contained in the two essential spray ponds meets the intent of this requirement.

The UHS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the ESPS to operate for at least 26 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ESPS. To meet this condition, the UHS temperature should not exceed 89°F and the level of each ESP should not fall below 12 ft usable water depth during normal unit operation. Since the bottom 1.5 ft of the ESPS is not usable, an actual depth of 13.5 ft provides a usable depth of 12 ft to meet the heat loads minimum water requirement.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, OPERABILITY requirements of the UHS are determined by the systems it supports.

Reference LCO 3.0.6 and the SFDP (Specification 5.5.15).

ACTIONS

A.1 and A.2

If the UHS is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate long term (26 days) cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the ESPS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the usable water level of each ESP is ≥ 12 ft. A usable water depth of 12 feet requires 13'-6" of actual water depth. The implementing procedure requires the operator to verify that the level is greater than or equal to 13'-6" measured locally at the spray pond or 14' indicated in the control room using installed instrumentation. The difference is a result of instrument uncertainty.

SR 3.7.9.2

This SR verifies that the ESPS is available to cool the EW System to at least its maximum design temperature within the maximum accident or normal design heat loads for 26 days following a DBA. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODEs. This SR verifies that the UHS water temperature is 89°F, as indicated in the control room. This value includes allowance for uncertainties.

REFERENCES

1. UFSAR, Section 9.2.5.
2. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.10 Essential Chilled Water (EC) System

BASES

BACKGROUND

The EC System provides a heat transfer system to the ultimate heat sink for the removal of process and operating heat from selected safety related air handling systems during a Design Basis Accident (DBA) or transient.

The EC System is a closed loop system consisting of two independent trains. Each 100% capacity train includes a heat exchanger, surge tank, pump, chemical addition tank, piping, valves, controls, and instrumentation. An independent 100% capacity chilled water refrigeration unit cools each train. The EC System is actuated on receipt of an ESFAS signal and supplies chilled water to the Heating, Ventilation, and Air Conditioning (HVAC) units in Engineered Safety Feature (ESF) equipment areas (e.g., the main control room, DC equipment room, AFW pump rooms, EW pump rooms and safety injection pump rooms).

The flow path for the EC System includes the closed loop of piping to all serviced equipment.

During normal operation, the normal Chilled Water System (WC) and the normal HVAC System cools the areas served by the EC System. The WC System and the normal HVAC System are nonsafety grade systems. Following ESFAS actuations, the EC System with essential HVAC units provide this cooling function to the control room and safety grade equipment. Additional information about the design and operation of the system, along with a list of components served, can be found in the UFSAR, Section 9.2.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the EC System is to remove the post accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. Each train provides chilled water to the HVAC units at the design temperature of 44°F. The design flowrate is 319 gpm for Train A and 314 gpm for Train B not including flow required for oil cooler.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The maximum heat load in the ESF pump room area occurs during the recirculation phase following a loss of coolant accident. During recirculation, hot fluid from the containment sump is supplied to the high pressure safety injection and containment spray pumps. This heat load to the area atmosphere must be removed by the EC System to ensure that these pumps remain OPERABLE.

The EC satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two EC trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single failure.

An EC train is considered OPERABLE when:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, refrigeration unit, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of the EC from other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EC System. The OPERABILITY of supported systems is determined in accordance with LCO 3.0.6 and the SFDP.

APPLICABILITY

In MODES 1, 2, 3, and 4, the EC System is required to be OPERABLE when a LOCA or other accident would require ESF operation.

In MODES 5 and 6, the OPERABILITY requirements of the ECS System are determined by the systems it supports. Reference LCO 3.0.6 and the SFDP (Specification 5.5.15).

(continued)

BASES

ACTIONS

A.1

If one EC train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this condition, one OPERABLE ECW train is adequate to perform the cooling function. The 72 hour Completion Time is reasonable, based on the low probability of an event occurring during this time and the 100% capacity OPERABLE EC train.

B.1 and B.2

If the EC train cannot be restored to OPERABLE status within the associated Completion Time, or two EC trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EC flow path provides assurance that the proper flow paths exist for EC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies proper automatic operation of the EC System components and that the EC pumps will start in the event of any accident or transient that generates an applicable ESFAS signal. This SR also ensures that each automatic valve in the flow paths actuates to its correct position on an actual or simulated ESFAS signal.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is based on operating experience and design reliability of the equipment.

REFERENCES

1. UFSAR, Section 9.2.9.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Essential Filtration System (CREFS)

BASES

BACKGROUND

The CREFS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodine), and a fan. Ductwork, dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines.

The CREFS is an emergency system. Upon receipt of the actuating signal(s), normal HVAC to the control room is isolated, and the stream of ventilation air is mixed with outside air and recirculated through the filter trains of the system. The prefilters remove any large particles in the air, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Actuation of CREFS aligns the system for recirculation of control room air mixed with outside air through the redundant trains of HEPA and charcoal filters. The CREFS initiates pressurization and filtered ventilation of the air supply to the control room.

(continued)

BASES

BACKGROUND (continued)

Outside air is filtered, and then added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building.

The air entering the control room is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the CREFS.

A single train is designed to pressurize the control room to ≥0.125 inches water gauge. The CREFS operation in maintaining the control room habitable is discussed in the FSAR, Section 6.4 (Ref. 1).

Redundant recirculation trains provide the required filtration. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.

The CREFS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREFS components are arranged in redundant safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access.

The CREFS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2).

Toxic gases are not stored or used onsite in quantities sufficient to necessitate control room protection, as required by Regulatory Guide 1.78 (Ref. 1). In addition, nearby industrial, military, and transportation facilities present no hazard to the operation of PVNGS, and there are no site-related design basis events due to accidents at these facilities (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a component of the CREFS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREFS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant trains of the CREFS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train. Total system failure could result in a control room operator receiving a dose in excess of 5 rem whole body or its equivalent in the event of a large radioactive release.

The CREFS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A CREFS train is considered OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREFS must be OPERABLE to limit operator exposure during and following a DBA.

In MODES 5 and 6, the CREFS is required to cope with the release from a rupture of a waste gas tank.

During movement of irradiated fuel assemblies, the CREFS must be OPERABLE to cope with the release from a fuel handling accident.

Refer to LCO 3.0.6 and the SFDP (Specification 5.5.15) to determine OPERABILITY requirements of supporting systems.

(continued)

BASES

ACTIONS

A.1

With one CREFS train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREFS subsystem is adequate to perform control room radiation protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

B.1 and B.2

If the inoperable CREFS cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

In MODE 5 or 6, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREFS train must be immediately placed in operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

D.1 and D.2

During movement of irradiated fuel assemblies, if required Action A.1 cannot be completed within the required Completion time, the OPERABLE CREFS train must be started immediately or movement of irradiated fuel assemblies must be suspended immediately. The first action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. If the system is not placed in operation, this action requires suspension of the movement of irradiated fuel assemblies in order to minimize the risk of a release of radioactivity that might require the actuation of CREFS. This does not preclude the movement of fuel to a safe position.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

When in MODES 5 and 6, or during movement of irradiated fuel assemblies with two CREFS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

F.1

If both CREFS trains are inoperable in MODE 1, 2, 3, or 4, the CREFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly operations for ≥ 15 minutes to demonstrate the function of the system is required. The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.

SR 3.7.11.2

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREFS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.3

This SR verifies each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is consistent with that specified in Reference 3.

SR 3.7.11.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREFS. During operation, the CREFS is designed to pressurize the control room ≥ 0.125 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure with one train at a ventilation flow rate of ≤1000 cfm. The ventilation flowrate is the outside makeup air flowrate. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800, Section 6.4 (Ref. 4).

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Chapter 15.
3. Regulatory Guide 1.52 (Rev. 2).
4. NUREG-0800, Section 6.4, Rev. 2, July 1981.
5. UFSAR, Section 9.4.
6. UFSAR, Section 2.2

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent, redundant trains that provide cooling of recirculated control room air. Each train consists of cooling coils, instrumentation, and controls to provide for control room temperature control. The CREATCS is a subsystem providing air temperature control for the control room.

The CREATCS is an emergency system, which is part of the Control Room Essential Filtration Sytem (CREFS). A single train will provide the required temperature control to maintain the control room between 70°F and 80°F. The CREATCS operation to maintain the control room temperature is discussed in the UFSAR, Section 9.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain temperature of the control room environment throughout 30 days of continuous occupancy.

The CREATCS components are arranged in redundant safety related trains. During emergency operation, the CREATCS maintains the temperature between 70°F and 80°F. A single active failure of a component of the CREATCS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREATCS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

Refer to LCO 3.0.6 and the SFDP (Specification 5.5.15) to determin the OPERABILITY requirements of suporting systems.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring control room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or nonsafety related cooling means that are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the required Completion Time, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

(continued)

BASES (continued)

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

In MODE 5 or 6, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

D.1 and D.2

During movement of irradiated fuel assemblies, if Required Action A.1 cannot be completed within the Required Completion Time, the OPERABLE CREATCS train must be started immediately or movement of irradiated fuel assemblies must be suspended immediately. The first action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. If the system is not placed in operation, this action requires suspension of the movement of irradiated fuel assemblies in order to minimize the risk of a release of radioactivity that might require the actuation of CREATCS. This does not preclude the movement of fuel to a safe position.

E.1 and E.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require actuation of CREATCS. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

F.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the CREATCS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. An 18 month Frequency is appropriate, since significant degradation of the CREATCS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.4.

B 3.7 PLANT SYSTEMS

B 3.7.13 Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ESF PREACS filters air from the area of the active ESF components during the recirculation phase of a Loss Of Coolant Accident (LOCA).

The ESF PREACS consists of two independent and redundant trains shared with the fuel building. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines and to back up the upstream HEPA filter, should it develop a leak. The system initiates filtered ventilation of the pump rooms and lower region of the auxiliary building following receipt of a safety injection actuation signal.

The ESF PREACS is a standby system. The Auxiliary Building Normal HVAC System provides normal cooling. During emergency operations, the ESF PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ESF pump rooms are isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ESF PREACS is discussed in the FSAR, Sections 6.5.1, 9.4.2, and 15.6.5 (Refs. 1, 2, and 3, respectively). The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ESF PREACS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as safety injection pump seal failure, during the recirculation mode. In such a case, the system limits the radioactive release to within 10 CFR 100 limits (Ref. 5). The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The ESF PREACS also actuates following a small break LOCA, requiring the unit to go into the recirculation mode of long term cooling and to clean up releases of smaller leaks, such as from valve stem packing.

The two types of system failures that are considered in the accident analysis are complete loss of function and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ESF envelope following a LOCA.

The ESF PREACS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant ESF PREACS trains are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ESF envelope exceeding the required limits in the event of a Design Basis Accident (DBA).

ESF PREACS is considered OPERABLE when the individual components necessary to maintain the ESF Pump Room filtration are OPERABLE in both trains.

An ESF PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

(continued)

BASES

LCO (continued)

c. Heater, prefilter, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the auxiliary building envelope below the 100 ft. elevation must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESF PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODES 5 and 6, the ESF PREACS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ESF PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ESF PREACS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1 and B.2

If the ESF PREACS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system.

Monthly operations for ≥ 15 minutes demonstrates the function of the system. There is not expected to be any moisture buildup on the adsorbers and HEPA filters due to the low humidity at PVNGS (Ref. 7). The 31 day Frequency is based on the known reliability of equipment, and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required ESF PREACS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ECCS PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.3

This SR verifies that each ESF PREACS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

SR 3.7.13.4

This SR verifies the integrity of the ESF envelope. The ability of the ESF envelope to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the ESF PREACS. During the post accident mode of operation, the ESF PREACS is designed to maintain a slight negative pressure in the ESF envelope with respect to adjacent areas to prevent unfiltered LEAKAGE. For the purposes of testing, the term

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.4 (continued)

"measurable negative pressure" is defined as 10 times the minimum instrument reading. The ESF PREACS is designed to maintain this negative pressure at a flow rate of 6,000 cfm ± 10% from the ESF envelope. The Frequency of 18 months is consistent with the guidance provided in the NUREG-0800, Section 6.5.1 (Ref. 6).

This test is conducted with the tests for filter penetration; thus, an 18 month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.2.
3. UFSAR, Section 15.6.5.
4. Regulatory Guide 1.52 (Rev. 2).
5. 10 CFR 100.11.
6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
7. UFSAR, Section 1.8

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2, Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the intent of the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is less than one-third of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the bundle and the surface, by the width of the bundle. The decontamination factor for 22 ft-6 in of water is essentially the same as that for 23 ft of water so the intent of Regulatory Guide 1.25 is met.

The fuel storage pool water level satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.9.6, "Refueling Water Level-Fuel Assemblies".

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.7.4.
4. Regulatory Guide 1.25
5. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

As described in LCO 3.7.17, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 2150 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron.

APPLICABLE SAFETY ANALYSES

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.17 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack or between a rack and the pool walls. These incidents could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by these postulated accident scenarios.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenario described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies.

(continued)

BASES

APPLICABILITY (continued)

With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1 and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, immediate action to perform a fuel storage pool verification can be initiated.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

1. UFSAR, Section 9.12.

2. PVNGS Operating License Amendments 82, 69 and 54 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated September 30, 1994.

B 3.7 PLANT SYSTEMS

B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.14, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.17, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity level, the resultant 2 hour thyroid dose to a person at the Exclusion Area Boundary (EAB) would be about .42 rem should the Main Steam Safety Valves (MSSVs) open for the 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through MSSVs and Atmospheric Dump Valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

(continued)

BASES

REFERENCES

1. 10 CFR 100.11.

2. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool was originally designed to store up to 1329 fuel assemblies in a borated fuel storage mode. The current storage configuration, which allows credit to be taken for burnup and does not require neutron absorbing (boraflex) storage cans, provides for a maximum storage of 1054 fuel assemblies in a three-region configuration. Region 1 is comprised of three 9x8 storage racks, one 12x8 storage rack and one 9x9 storage rack. To prevent inadvertent storage of a fuel assembly in a cell required to be vacant, cell blocking devices are placed in every other storage cell location to maintain a two-out-of-four checkerboard configuration. Region 2 is comprised of three 9x8 storage racks and one 12x8 storage rack. Cell blocking devices in Region 2 are employed in one out of every four storage cell locations to preclude the possibility of an unanalyzed assembly configuration. Region 3 is comprised of six 9x8 storage racks and two 12x8 storage racks. Since fuel assemblies may be stored in every Region 3 cell location, no cell blocking devices are installed in Region 3. Cell blocking devices are also placed along the Region 2 interface with Region 3 to eliminate the possibility of an unanalyzed arrangement of assemblies. The spent fuel storage cells are installed in parallel rows with a nominal center-to-center spacing of 9.5 inches. This spacing and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figure 3.7.17-1 is sufficient to maintain a k_{eff} of $\leq$0.95 for spent fuel of original enrichment of up to 4.30%.

APPLICABLE SAFETY ANALYSES

The spent fuel storage pool is designed for non-criticality by use of adequate spacing, and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figure 3.7.17-1.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed by ABB-Combustion Engineering (CE) using the two-dimensional discrete ordinates transport theory DOT-IV computer code, with four energy group neutron cross sections generated by the CEPAK code. These codes have been previously used by CE for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the PVNGS fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing. In March 1992, the NRC issued Information Notice 92-21 and Supplement 1 concerning discrepancies that were discovered in spent fuel pool reactivity calculations. The discrepancies were due to an overestimation of neutron absorption in the CEPAK generation of cross sections. These discrepancies were found to exist only in regions containing a strong neutron absorber (poison). Since neutron poison is not present, this problem does not exist for the PVNGS racks.

The modeling of Regions 2 and 3 included several conservative assumptions. These assumptions neglected the reactivity effects of axial leakage, poison shims in the assemblies, structural grids, and soluble boron in the 68°F pool water. These assumptions tend to increase the calculated effective multiplication factor (k_{eff}) of the racks. The stored fuel assemblies were modeled as CE 16x16 assemblies with a nominal pitch of 0.506 inches between fuel rods, a fuel pellet diameter of 0.33 inches, and a UO(2) density of 10.4 g/cc.

DOT-IV calculations were used to construct a curve of burnup versus initial enrichment for both Regions 2 and 3 (TS Figure 5.6-1) such that all points on the curve produce a k_{eff} value (without uncertainties or biases) of 0.93. This method of reactivity equivalencing has been accepted by the NRC and used for numerous other spent fuel storage pools which take credit for burnup. The NRC criticality acceptance criterion for fuel storage is that k_{eff} be no greater than 0.95, including all uncertainties at a 95% probability/95% confidence level. Therefore, the reactivity of assemblies, minimum monolith thickness, temperature variations, minimum L-insert thickness, assembly enrichment, and assembly burnup were obtained as well as a methodology uncertainty and bias. These were applied to the nominal value of 0.93 to obtain a final k_{eff} 0.944 for the spent fuel racks. This meets the NRC criterion of no greater than 0.95.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as an assembly drop on top of a rack or between a rack and the pool walls or the misloading of an assembly, with a burnup and enrichment combination outside of the acceptable area in TS Figure 3.7.17-1, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of 2150 ppm of boron in the pool water required by TS 3.7.15 since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k_{eff} caused by the boron more offsets the reactivity addition caused by credible accidents. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

The criticality aspects of the spent fuel pool meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

The spent fuel pool head load calculations were based on a full pool with 1300 fuel assemblies. The maximum number of fuel assemblies that can be stored in the three-region configuration is 1054 fuel assemblies. The actual loading pattern therefore has a lower decay heat than assumed in the calculations for a full pool.

The original licensing basis for the spent fuel pool allowed for spent fuel to be loaded in either a 4x4 array or a checkerboard array, depending on the use of boraflex poison. Therefore, a fuel handling accident was assumed to occur with maximum loading of the pool. The fuel pool rack construction precludes more than one assembly from being impacted in a fuel handling accident. Therefore, the UFSAR analysis conclusion regarding the worst scenario for a dropped assembly (in which the horizontal impact of a fuel assembly on top of the spent fuel assembly damages fuel rods in the dropped assembly but does not impact fuel in the stored assemblies) continued to be limiting.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 3.7.17-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Figure 3.7.17-1, in the accompanying LCO. Specification 4.3.1.1 provides additional details for fuel storage in each of the three Regions.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.17-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.17-1.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 in the accompanying LCO and Specification 4.3.1.1.

(continued)

BASES

REFERENCES

1. UFSAR, Section 9.1.2.

2. PVNGS Operating License Amendments 82, 69, and 54 for Units 1, 2, and 3 respectively, and associated NRC Safety Evaluation, dated September 30, 1994.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources – Operating

BASES

BACKGROUND

The unit Class 1E Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources: normal and alternate(s)), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Features (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources (normal and alternate) and a single DG.

Offsite power is supplied to the unit switchyard from the transmission network by five transmission lines. From the switchyard, two electrically and physically separated circuits provide AC power, through ESF service transformers, to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the updated FSAR, Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer (NBN-X03 and NBN-X04) supplying offsite power to the onsite Class 1E Distribution System. Within 30 seconds after the initiating signal is received, all permanently connected and auto-connected emergency loads needed to recover the unit or maintain it in a safe condition are returned to service via the automatic load sequencer.

(continued)

BASES

BACKGROUND (continued)

The onsite standby power source for each 4.16 Kv ESF bus is a dedicated DG. DG-A and DG-B are dedicated to ESF buses PBA-S03 and PBB-S04, respectively. A DG starts automatically (in emergency mode) on a safety injection actuation signal (SIAS) (i.e., low pressurizer pressure or high containment pressure signals), auxiliary feedwater actuation signals (AFAS-1 and AFAS-2) (e.g., low steam generator level), or on a loss of power (an ESF bus degraded voltage or undervoltage signal). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with a SIAS or AFAS signal. Following the loss of offsite power, the sequencer sheds nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application. The DGs will also start and operate in the standby mode (running unloaded) without tying to the ESF bus on a SIAS or AFAS.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within 40 seconds after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 5500 Kw with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 Kv ESF buses are listed in the updated FSAR, Chapter 8 (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the updated FSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Offsite circuits are those that are described in the updated FSAR and are part of the licensing basis for the unit.

In addition, one automatic load sequencer per train must be OPERABLE.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

(continued)

BASES

LCO (continued)

The startup transformers (NAN-X01, NAN-X02, and NAN-X03) convert the 525 Kv offsite power to the Non-Class 1E 13.8 Kv power. Each secondary winding of a startup transformer normally provides power to one of two interconnected 13.8 Kv intermediate buses (NAN-S05 & NAN-S06) per unit, in such a way that the two 13.8 Kv intermediate buses of the same unit receive power from two different start-up transformers (preferred offsite sources: normal and alternate supply). For example, Unit 1 NAN-S05's normal supply is from a NAN-X03 secondary winding and NAN-S05's alternate supply is from a NAN-X01 secondary winding; Unit 1 NAN-S06's normal supply is from a NAN-X02 secondary winding and NAN-S05's alternate supply is from a NAN-X01 secondary winding. The secondary winding are sized to start and carry one-half of the non-Class 1E loads of one unit and two trains of ESF loads, one which is from another unit, during unit trips or during startup/shutdown operation.

The 13.8 Kv intermediate buses (NAN-S05 & NAN-S06), in turn, distribute power to the 4.16 kV Class 1E buses (PBA-S03 & PBB-S04) via a 13.8 kV bus (NAN-S03 or NAN-S04) and an ESF transformer (NBN-X03 or NBN-X04).

Two fast bus transfer circuits are also provided to transfer the non-Class 1E house loads fed from NAN-S01 and NAN-S02 to 13.8 kV buses NAN-S03 and NAN-S04 respectively during a plant trip or during startup/shutdown operation. Prior to a plant trip, NAN-S01 and NAN-S02 are fed from the auxiliary transformer, and are fed from NAN-S03 and NAN-S04 respectively after the plant trip.

Each DG must be capable of starting, accelerating to rated speed (i.e., frequency) and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within (≤) 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby condition with the engine hot and DG in standby condition with the engine at normal keep-warm conditions. Additional DG capabilities must be demonstrated to meet required Surveillances (e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode).

(continued)

BASES

LCO (continued)

Proper sequencing of loads, including tripping of nonessential loads, is a required function.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

APPLICABILITY

The AC sources and sequencers are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are covered in LCO 3.8.2, "AC Sources – Shutdown."

(continued)

BASES

ACTIONS

A.1

To ensure a highly reliable power source remains with the one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the train (i.e., ESF bus) cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features require Class 1E power from PBA-S03 or PBB-S04 ESF buses to be OPERABLE, and include: charging pumps; radiation monitors Train A RU-29 and Train B RU-30 (TS 3.3.9), Train A RU-31 and Train B RU-145; pressurizer heaters (TS 3.4.9); ECCS (TS 3.5.3 and TS 3.5.4); containment spray (TS 3.6.6); containment isolation valves NCA-UV-402, NCB-UV-403, WCA-UV-62, and WCB-UV-61 (TS 3.6.3); containment hydrogen monitors (TS 3.3.10); hydrogen recombiners (TS 3.6.7); auxiliary feedwater system (TS 3.7.5); essential cooling water system (TS 3.7.7); essential spray pond system (TS 3.7.8); essential chilled water system (TS 3.7.10); control room essential filtration system (TS 3.7.11); control room emergency air temperature control system (TS 3.7.12); ESF pump room air exhaust cleanup system (TS 3.7.13); shutdown cooling subsystems (TS 3.4.6, 3.6.7, 3.6.8, and 3.4.15); and fuel building ventilation. Mode applicability is as specified in each appropriate TS section.

(continued)

BASES

ACTIONS

A.2 (continued)

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads; and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed

(continued)

BASES

ACTIONS

A.3 (continued)

72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable, and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 6 day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

(continued)

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if an offsite circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of redundant required features. These features require Class 1E power from PBA-S03 or PBB-S04 ESF buses to be OPERABLE, and are identical to those specified in ACTION A.2. Mode applicability is as specified in each appropriate TS section. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action.

(continued)

BASES

ACTIONS

B.2 (continued)

Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other DG, the other DG would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

(continued)

BASES

ACTIONS (continued)

B.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 6 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. These features require Class 1E power from PBA-S03 or PBB-S04 ESF buses to be OPERABLE, and are identical to those specified in ACTION A.2. Mode applicability is as specified in each appropriate TS section.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Regulatory Guide 1.93 (Ref. 6), with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to a train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems – Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Regulatory Guide 1.93 (Ref. 6), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

The sequencer(s) is an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. Therefore, loss of an ESF bus sequencer affects every major ESF system in the load group. The 24 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal. Required Action F.2 is intended to provide assurance that a single failure of a DG Sequencer will not result in a complete loss of safety function of critical redundant required features.

G.1, G.2.1, G.2.2.1, and G.2.2.2

To ensure adequate voltage is delivered to downstream electrical equipment, the Electrical Distribution System (EDS) requires an adequate input voltage. Failure to assure adequate input voltage, may result in double sequencing should an accident requiring sequencer operation and a Fast Bus Transfer (FBT) occur during the period of low input voltage to the EDS.

Adequate EDS voltage is assured by maintenance of switchyard voltage at or above 524 kV for Unit 1, and 518 kV for Units 2 and 3, when all three startup transformers are in service. When only two startup transformers are in service, adequate voltage is assured to all three Units by maintaining switchyard voltage at or above 525 kV. Blocking FBT ensures adequate EDS voltage by preventing the non-Class 1E House Loads from depressing the Class 1E 4.16 kV switchgear voltage.

Action G.2.1 applies only if switchgear breakers NAN-S03B and NAN-S04B are open (i.e, FBT has not yet occurred). If breakers NAN-S03B and NAN-S04B are closed (i.e., FBT has occurred) and the degraded voltage relays (DVRs) have not activated, then the LCO is met, regardless of switchyard voltage, and no action is required as the DVRs are capable

(continued)

BASES

ACTIONS (continued)

of determining adequate voltage at the Class 1E 4.16 kV buses. Also, if the switchyard voltage perturbation is less than 28.62 seconds, then the LCO is met and no action is required as the perturbation is less than the current, minimum time delay of the DVRs.

Starting, loading and separating the opposite train's EDG from offsite power prevents FBT from depressing the Class 1E 4.16 kV bus voltage. In addition, this configuration provides offsite power to half of the House Loads to provide forced circulation in the event of a plant trip.

Action G.2.2.1 applies only if switchgear breakers NAN-S03B and NAN-S04B are open (i.e, FBT has not yet occurred). If breakers NAN-S03B and NAN-S04B are closed (i.e., FBT has occurred) and the degraded voltage relays (DVRs) have not activated, then the LCO is met, regardless of switchyard voltage, and no action is required as the DVRs are capable of determining adequate voltage at the Class 1E 4.16 kV buses. Also, if the switchyard voltage perturbation is less than 28.62 seconds, then the LCO is met and no action is required as the perturbation is less than the current, minimum time delay of the DVRs.

(continued)

BASES

ACTIONS
(continued)

The 72 hour completion time establishes a limit on the maximum time allowed to restore adequate Electrical Distribution System (EDS) input voltage. Regulatory Guide 1.93 (Ref. 6) recognizes that under certain conditions it may be safer to continue operation at full or reduced power for a limited time than to effect an immediate shutdown based on the loss of some of the required electric power sources. Action G balances the risk of a forced shutdown against the risk of remaining at power with EDS input voltage restored by blocking Fast Bus Transfer or starting, loading, and separating EDGs from offsite power. Action G.2.1 maintains the availability of all four electric power sources, however, House Loads would not be available for forced circulation capability during plant trips. Action G.2.2.1 reduces the number of available electric power sources and maintains partial forced circulation capability.

H.1 and H.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions).

The SR for demonstrating OPERABILITY of the DGs are based on the recommendations of Regulatory Guide 1.9 (Ref. 3), unless otherwise noted in the Updated FSAR Section 1.8.

The DG capabilities (starting and loading) are required to be met from a variety of initial conditions such as DG in standby condition with the engine hot (SR 3.8.1.15) and DG in standby condition with the engine at normal keep-warm conditions (SR 3.8.1.2, SR 3.8.1.7 and SR 3.8.1.19). Although it is expected that most DG starts will be performed from normal keep-warm conditions, DG starts should be performed with the jacket water cooling and lube oil temperatures within the lower to upper limits of DG OPERABILITY, except as noted above. Rapid cooling of the DG down to normal keep-warm conditions should be minimized.

The required steady state frequency range for the DG is 60 +1.2/-0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow. In accordance with the guidance provided in Regulatory Guide 1.9 (Ref. 3), where steady state conditions do not exist (i.e., transients), the frequency range should be restored to within ± 2% of the 60 Hz nominal frequency (58.8 Hz to 61.2 Hz).

Specific MODE restraints have been footnoted where applicable to each 18 month SR. The reason for "This Surveillance shall not be performed in MODE 1 or 2" is that during operation with the reactor critical, performance of this SR could cause perturbations to the EDS that could challenge continued steady state operation and, as a result, unit safety systems; or that performing the SR would remove a required DG from service. The reason for "This Surveillance shall not be performed in MODE 1, 2, 3, or 4" is that performing this SR would remove a required offsite circuit from service, perturb the EDS, and challenge safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1

This SR assures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and indicated availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby condition. Standby conditions for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, the DG manufacturer recommends a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

SR 3.8.1.7 requires that, at a 184 day Frequency, the DG starts from standby conditions with the engine at normal keep-warm conditions and achieves required voltage and frequency within 10 seconds, and subsequently achieves steady state required voltage and frequency ranges. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter 15 (Ref. 5).

A minimum voltage and frequency is specified rather than an upper and a lower limit because a diesel engine acceleration at full fuel (such as during a fast start) is likely to "overshoot" the upper limit initially and then go through several oscillations prior to a voltage and frequency within the stated upper and lower bounds. The time to reach "steady state" could exceed 10 seconds, and be cause to fail the SR. However, on an actual emergency start, the EDG would reach minimum voltage and frequency in $\leq$ 10 seconds at which time it would be loaded. Application of the load will dampen the oscillations. Therefore, only specifying the minimum voltage and frequency (at which the EDG can accept load) demonstrates the necessary capability of the EDG to satisfy safety requirements without including a potential for failing the Surveillance.

While reaching minimum voltage and frequency in $\leq$ 10 seconds is an immediate test of OPERABILITY, the ability of the governor and voltage regulator to achieve steady state operation, and the time to do so are important indicators of continued OPERABILITY. Therefore, the time to achieve steady state voltage and frequency will be monitored as a function of continued OPERABILITY.

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, 10 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

The normal 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads of 90 to 100 percent (4950 - 5500 kW) of the continuous rating of the DG. Consistent with the guidance provided in the Regulatory Guide 1.9 (Ref. 3) load-run test description, the 4950 - 5500 kW band will demonstrate 90 to 100 percent of the continuous rating of the DG. The load band (4950 - 5500 kW) is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing may be performed within the guidance of the generator capability curve.

A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

The normal 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 9). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

Since the design of the fuel transfer system is such that pumps will operate automatically in order to maintain an adequate volume of fuel oil in the day tank during or following DG testing, a 31 day Frequency is appropriate.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the auto-connected emergency loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load, or equivalent load, without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. Train A Normal Water Chiller (at 842 kW) and Train B AFW pump (at 904 kW) are the bounding loads for the DG A and DG B to reject, respectively. These values are listed in Table 8.3-3, Load Bases for Class 1E Buses, in the Updated FSAR (Ref. 2). This Surveillance may be accomplished by:

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. 12), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.9 (Ref. 3).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

This SR is modified by a Note. The reason for the Note is that performing this SR would remove a required offsite circuit from service, perturb the EDS, and challenge safety systems. This SR is performed in emergency mode (not paralleled to the grid) ensuring that the DG is tested under load conditions that are as close to design basis conditions as possible.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing is performed using design basis kW loading and maximum kVAR loading permitted during testing. These loads represent the inductive loading that the DG would experience to the extent practicable and is consistent with the guidance of Regulatory Guide 1.9 (Ref. 3). Consistent with the guidance provided in the Regulatory Guide 1.9 full-load rejection test description, the 4950 - 5500 kW band will demonstrate the DG's capability to reject a load equal to 90 to 100 percent of its continuous rating. Administrative limits have been placed upon the Class 1E 4160 V buses due to high voltage concerns. As a result, power factors deviating much from unity are currently not possible when the DG runs parallel to the grid. To the extent practicable, VARs will be provided by the DG during this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.8 (Ref. 3) and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

SR 3.8.1.11

As required by Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.4, this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG auto-start time of 10 seconds is derived from requirements of the accident analysis. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected emergency loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or shutdown cooling (SDC) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified to the extent possible ensuring power is available to the component.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Note 3 states that momentary voltage and frequency transients induced by load changes do not invalidate this test.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis accident signal (LOCA) signal, and subsequently achieves steady state required voltage and frequency ranges, and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and auto-connected emergency loads (auto-connected through the automatic load sequencer) are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

The requirement to verify the connection of permanent and auto-connected emergency loads is intended to satisfactorily show the relationship of these loads to the offsite circuit loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

capable of being operated at full flow, or SDC systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the offsite circuit system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified to the extent possible ensuring power is available to the component.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. The reason for Note 2 is that performing this SR would remove a required offsite circuit from service, perturb the EDS, and challenge safety systems.

SR 3.8.1.13

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal, and critical protective functions (engine overspeed, generator differential current, engine low lube oil pressure, and manual emergency stop trip), trip the DG to avert substantial damage to the DG unit. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service.

SR 3.8.1.14

Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.9, requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent to 105 to 110% of the continuous rating of the DG (5775 - 6050 kW) and ≥ 22 hours at a load equivalent to 90 to 100% of the continuous duty rating of the DG (4950 - 5500 kW). The DG starts for this Surveillance can be performed either from normal keep-warm or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR (Note 3 and Note 4).

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing is performed using design basis kW loading and maximum kVAR loading permitted during testing. These loads represent the inductive loading that the DG would experience to the extent practicable and is consistent with the intent of Regulatory Guide 1.9 (Ref. 3). Administrative limits have been placed upon the Class 1E 4160 V buses due to high voltage concerns. As a result, power factors deviating much from unity are currently not possible when the DG runs parallel to the grid. To the extent practicable, VARs will be provided by the DG during this SR. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.9, takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by four Notes. Note 1 states that momentary variations due to changing bus loads do not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR (Note 3 and Note 4).

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds, and subsequently achieves steady state required voltage and frequency ranges. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.10.

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. Per the guidance in Regulatory Guide 1.9, this SR would demonstrate the hot restart functional capability at full-load temperature conditions, after the DG has operated for 2 hours (or until operating temperatures have stabilized) at full load. Momentary transients due to changing bus loads do not invalidate the test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16

As required by Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.11, this Surveillance ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, in standby operation (running unloaded), the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready-to-load operation if a LOCA actuation signal (e.g., simulated SIAS) is received during operation in the test mode. Ready-to-load operation is defined as the DG running at rated speed and voltage, in standby operation (running unloaded) with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 12), paragraph 6.2.6(2) and Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.13.

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.18

Under accident and loss of offsite power conditions loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The 1 second load sequence time tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. FSAR, Chapter 8 (Ref. 2) provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), paragraph 2.2.4; takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), paragraph 2.3.2.4 and Regulatory Guide 1.137 (Ref. 9).

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. The reason for Note 2 is that during operation with the reactor critical, performance of this SR could cause perturbations to the EDS that could challenge continued steady state operation and, as a result, unit safety systems.

REFERENCES:

1. 10 CFR 50, Appendix A, GDC 17
2. Updated FSAR, Chapter 8
3. Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993.
4. Updated FSAR, Chapter 6
5. Updated FSAR, Chapter 15
6. Regulatory Guide 1.93, "Availability of Electric Power Sources," Revision 0, December 1974.
7. GL 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18
9. Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.
10. ANSI C84.1-1982
11. IEEE Standard 308-1974, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources – Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and minimal in consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed

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BASES

APPLICABLE SAFETY ANALYSES (continued)

within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems – Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the

(continued)

BASES

LCO (continued)

offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Offsite circuits are those that are described in the updated FSAR and are part of the licensing basis for the unit. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of the offsite circuit.

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby condition with the engine hot and DG in standby condition at normal keep-warm conditions.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

(continued)

BASES

APPLICABILITY (continued)

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

The ACTIONS are modified by a Note that identifies required Actions A.2.3 and B.3 are not applicable to the movement of irradiated fuel assemblies in Modes 1 through 4.

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains may be required by LCO 3.8.10, the remaining train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4
(continued)

additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, inability to immediately suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. The SRs that are applicable and required to be performed are SR 3.8.1.1, SR 3.8.1.2, SR 3.8.1.4, SR 3.8.1.5, and SR 3.8.1.7. The SRs listed in the Note are not required to be performed as a condition of OPERABILITY because their performance would unnecessarily challenge the only remaining OPERABLE DG or offsite circuit. In addition, SR 3.8.1.6 is not required to be performed since the fuel oil transfer pump would not cycle without the one-hour load demand SR or the 24-hour run SR, neither of which is required to be performed.

The reasons for the exception to SR 3.8.2.1 applicability are as follows: SR 3.8.1.8 is not applicable since only one offsite circuit is required to be OPERABLE and an alternate offsite circuit may not be available; SR 3.8.1.12, SR 3.8.1.17, and SR 3.8.1.19 are not applicable because the ESF functions (i.e., AFAS and SIAS) are not required to be OPERABLE during shutdown; SR 3.8.1.17 is not applicable because the required OPERABLE DG(s) is not required to undergo periods of being load tested (parallel to the offsite circuit). SR 3.8.1.20 is not applicable because starting independence is not required with DG(s) that are not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC Sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days, while the DG is supplying maximum post loss of coolant accident load demand as discussed in the FSAR, Section 9.5.4.2.1 (Ref. 1). The maximum load demand is calculated using the assumption that at least two DGs are available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by a transfer pump associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All outside tanks, pumps, and piping are located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of 7 days of operation. This supply is sufficient supply to allow the operator to replenish lube oil from outside sources.

Each DG has independent and redundant starting air subsystems. Each DG starting subsystem provides a stored compressed air supply sufficient for accomplishing a DG start in $\leq$ 10 seconds. Each air receiver has been sized to accomplish 5 consecutive DG starts from the receiver design working pressure without being refilled.

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BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 4), and in the FSAR, Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources—Operating," and LCO 3.8.2, "AC Sources—Shutdown."

The starting air system is required to have a minimum capacity for five consecutive DG start attempts without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and starting air subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil and starting

(continued)

BASES

APPLICABILITY (continued)

air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition (i.e., < 80% indicated fuel level), the 7 day fuel oil supply (68,900 gallon of fuel) for a DG is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a 6 day supply (59,800 gallons of fuel). These circumstances may be caused by events such as full load operation required after an inadvertent start while at minimum required level; or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (≥ 6 days or ≥ 71% indicated fuel level), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

With lube oil inventory < 2.5 inches visible in the sightglass, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply.

(continued)

BASES

ACTIONS

B.1 (continued)

This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

The normal level of lube oil is maintained at mid-scale visible on the sightglass which ensures sufficient lube oil to support at least 13.5 days of engine operation during periods when the DG is supplying maximum post-LOCA load demand as discussed in the FSAR (Ref. 1). This is based on a conservative lube oil consumption rate of 1.5 gallons per hour and 486 gallons of available lube oil between the top of the lube oil suction pipe in the engine crankcase (minimum available level) and the mid-scale position on the sightglass. 252 gallons or 7 days of available lube oil is actually indicated at 1 inch visible in the sightglass. With ≥ 2.5 inches visible in the sightglass, a conservative supply of lube oil is ensured for 7 days of full load operation.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

(continued)

BASES

ACTIONS (continued)

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

Each DG is OPERABLE with one air receiver capable of delivering an operating pressure of ≥ 230 psig indicated. Although there exist two independent and redundant starting air receivers per DG, only one starting air receiver is required for DG OPERABILITY. Each receiver is sized to accomplish 5 DG starts from its normal operating pressure of 250 psig, and each will start the DG in ≤ 10 seconds with a minimum pressure of 185 psig. If the required starting air receiver is < 230 psig and ≥ 185 psig, the starting air system is degraded and a period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This 48-hour period is acceptable based on the minimum starting air capacity (≥ 185 psig), the fact that the DG start must be accomplished on the first attempt (there are no sequencial starts in emergency mode), and the low probability of an event during this brief period. Calculation 13-JC-DG-203 (Ref. 9) supports the proposed values for receiver pressures.

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The 2.5 inches visible in the sightglass requirement is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level.

The 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the fuel oil in accordance with ASTM-D4057 (Ref. 6);

b. Verify in accordance with the tests specified in ASTM D975 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89, or an API gravity at 60°F of ≥ 27° and ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F; and

c. Verify in accordance with the tests specified in ASTM D1796 (Ref. 6) that the sample water and sediment is ≤ 0.05 percent volume.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975 (Ref. 6), except that the analysis for cetane number may be performed in accordance with ASTM D976 (Ref. 6) or ASTM D4737 (Ref. 6). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

Particulate concentrations should be determined in accordance with ASTM D2276, Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank must be considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds or cranking) or engine cranking speed. The pressure specified in this SR is intended to reflect the lowest value at which the DG can be considered OPERABLE.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.5 (continued)

fuel oil storage tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of this Surveillance.

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. ASTM Standards: D4057-81; D975-91; D976-91; D4737-90; D1796-83; D2276-89, Method A.
7. ASTM Standards, D975, Table 1.
8. ASME, Boiler and Pressure Vessel Code, Section XI.
9. "Emergency Diesel Generator and Diesel Fuel Oil Systems Instrumentation Uncertainty Calculation", 13-JC-DG-203, Parts 23 and 51

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources – Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital instrument bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A and Train B). Each subsystem consists of two 125 VDC batteries (each battery ≥ 100% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling. Each subsystem contains two DC power channels. There are four channels designated as A and C for Train A, and B and D for Train B for each unit (See 3.8.4 LCO Bases section for detailed description).

Additionally there is one backup battery charger per subsystem, which provides backup service in the event that the normal battery charger is out of service. If the backup battery charger is substituted for one of the normal battery chargers, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, the 125 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The Train A and Train B DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital instrument buses.

(continued)

BASES

BACKGROUND (continued)

The DC power distribution system is described in more detail in the Bases for LCO 3.8.9, "Distribution Systems-Operating," and for LCO 3.8.10, "Distribution Systems – Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours as discussed in the UFSAR, Chapter 8 (Ref. 4).

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

In addition, each DC electrical power subsystem contains a backup battery charger which is manually transferable to either channel of a subsystem. The transfer mechanism is mechanically interlocked to prevent both DC channels of a subsystem from being simultaneously connected to the backup battery charger.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% (low specific gravity cells) or 90% (AT&T) of nameplate rating. The voltage limit is 2.13 V per cell (low specific gravity cells) or 2.18 V per cell (AT&T), which corresponds to a total minimum voltage output of 128 V per battery (low specific gravity cells) or 131 V per battery (AT&T) discussed in the Design Basis Manual (Ref. 12).

Each Train A and Train B DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 4).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 6) and Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The DC electrical power subsystems, each subsystem consisting of two batteries, battery charger for each battery (the backup battery charger, one per train, may be used to satisfy this requirement), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

Each DC electrical power subsystem (Train A or Train B) is subdivided into channels. Train A consists of Channel A and Channel C. Train B consists of Channel B and Channel D.
Channel A includes 125 VDC bus PKA-M41, 125 VDC battery bank PKA-F11, and normal battery charger PKA-H11 or backup battery charger PKA-H15.
Channel C includes 125 VDC bus PKC-M43, 125 VDC battery bank PKC-F13, and normal battery charger PKC-H13 or backup battery charger PKA-H15.

(continued)

BASES

Channel B includes 125 VDC bus PKB-M42, 125 VDC battery bank PKB-F12, and normal battery charger PKB-H12 or backup battery charger PKB-H16.
Channel D includes 125 VDC bus PKD-M44, 125 VDC battery bank PKD-F14, and normal battery charger PKD-H14 or backup battery charger PKB-H16.

An OPERABLE DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es).

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in the Bases for LCO 3.8.5, "DC Sources – Shutdown."

ACTIONS

A.1

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. This condition is independent of the battery charger status. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required DC electrical power subsystems is inoperable (exclusive of the battery charger), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the complete loss of the remaining 125 VDC electrical

(continued)

power subsystem with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1 and B.2

If the inoperable DC electrical power subsystem (exclusive of the battery charger) cannot be restored to OPERABLE status within the required Completion Time of Condition A, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).

C.1 and C.2

Condition C represents the loss of the required (in-service) battery charger and assumes that action will be taken immediately to restore charging capability to the battery with the alternate charger (i.e., normal or backup). Under normal plant load conditions, the loss of the battery charger for $\leq$ 1 hour has a negligible effect on the rated battery capacity and does not impact the DC electrical power subsystem's capability to perform its DBA safety function. Immediately following the loss of the charging capability, battery cell parameters may not meet Category A limits because these limits assume that the battery is being charged at a minimum float voltage. The 1 hour Completion Time allows for re-establishing charging capability such that Category A parameters can be met. Operation with the DC electrical power subsystem battery charger inoperable is not allowed for an indefinite period of time even when the battery cell parameters have been verified to meet the Category A limits of Table 3.8.6-1. The 24 hour completion time provides a period of time to correct the problem commensurate with the importance of maintaining the DC electrical power subsystem battery charger in an OPERABLE status.

(continued)

BASES

D.1

If the battery cell parameters cannot be maintained within Category A limits as specified in LCO 3.8.6, the short term capability of the battery is also degraded and the battery must be declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculation. The 7 day Frequency is more conservative than the frequency specified in IEEE-450 (Ref. 9).

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each inter-cell, inter-rack, inter-tier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The limits established for this SR are based on calculation 1,2,3ECPK207 which states that if every terminal connection were to degrade to 150E-6 ohms, there would be sufficient battery capacity to satisfy the DBA Duty Cycle (Ref. 13).

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

This SR is consistent with IEEE-450 (Ref. 9), which recommends detailed visual inspection of cell condition and rack integrity. The 18 month Surveillance Frequency is consistent with expected fuel cycle length, minimizing battery testing while on line that could result in rendering the batteries inoperable.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

The connection resistance limits for SR 3.8.4.5 is based on calculation 1,2,3ECPK207 which states that if every terminal connection were to degrade to 150E-6 ohms there would be sufficient battery capacity to satisfy the DBA Duty Cycle (Ref. 13).

The Surveillances are consistent with IEEE-450 (Ref. 9), which recommends cell to cell and terminal connection resistance measurement. The 18 month Surveillance Frequency is consistent with expected fuel cycle length, minimizing battery testing while on line that could result in rendering the batteries inoperable.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.6

This SR requires that each required battery charger be capable of supplying 400 amps for batteries A and B and 300 amps for batteries C and D, and 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle length.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance on the charger credited for OPERABILITY would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10) and Regulatory Guide 1.129 (Ref. 11), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the performance of a battery performance discharge test or a modified performance discharge test in SR 3.8.4.8 in lieu of a service test since both performance discharge test parameters envelope the service test. The reason for Note 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the "as found" condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The modified performance discharge test is a simulated duty cycle consisting of just two rates: the one minute rate published for the battery or the largest current load of the duty cycle (but in no case lower than the performance test rate), followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8. In addition, either of the performance discharge tests may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time, because the test parameters envelope the service test described in SR 3.8.4.7.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

capacity of 80% (low specific gravity cells) or 90% (AT&T) shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated when the battery capacity drops by more than 10% (low specific gravity cells) or 5% (AT&T) relative to its capacity on the previous performance test, or when it is ≥ 10% (low specific gravity cells) or ≥ 5% (AT&T) below the manufacturer's rating.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

REFERENCES

1. 10 CFR.50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1974.
4. UFSAR, Chapter 8.3.2.
5. IEEE-485-1983, June 1983.
6. UFSAR, Chapter 6.
7. UFSAR, Chapter 15.
8. Regulatory Guide 1.93, December 1974.
9. IEEE-450-1980.

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BASES

REFERENCES (continued)

10. Regulatory Guide 1.32, Revision 0, August 11, 1972.

11. Regulatory Guide 1.129, Revision 1, February 1978.

12. Design Basis Manual "Class 1E 125 VDC Power System".

13. Calculation 1,2,3ECPK207

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources – Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The DC electrical power subsystems, each subsystem consisting of two batteries, one battery charger per battery and the corresponding control equipment and interconnecting cabling within the train, are required to

(continued)

BASES

LCO (continued)

be OPERABLE to support required trains of distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems – Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

a. Required features needed to mitigate a fuel handling accident are available;

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

The Actions are modified by a Note that identifies required Action A.2.3 is not applicable to the movement of irradiated fuel assemblies in Modes 1 through 4.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two 125 VDC trains are required per LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts.

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, inability to immediately suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

B.1 and B.2

Condition B represents the loss of the required (in-service) battery charger and assumes that action will be taken immediately to restore charging capability to the battery with the alternate charger (i.e., normal or backup). Under normal plant load conditions, the loss of the battery charger for $\leq$ 1 hour has a negligible effect on the rated battery capacity and does not impact the DC electrical power subsystem's capability to perform its DBA safety function. Immediately following the loss of the charging capability, battery cell parameters may not meet Category A limits because these limits assume that the battery is being charged at a minimum float voltage. The 1 hour Completion

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Time allows for re-establishing charging capability such that Category A parameters can be met. Operation with the DC electrical power subsystem battery charger inoperable is not allowed for an indefinite period of time even when the battery cell parameters have been verified to meet the category A limits of Table 3.8.6-1. The 24 hours completion time provides a period of time to correct the problem commensurate with the importance of maintaining the DC electrical power subsystem battery charger in an OPERABLE status.

C.1

If the battery cell parameters cannot be maintained within Category A limits as specified in LCO 3.8.6, the short term capability of the battery is also degraded and the battery must be declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 states that Surveillances required by SR 3.8.4.1 through SR 3.8.4.8 are applicable in these MODES. See the corresponding Bases for LCO 3.8.4 for a discussion of each SR. This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources – Operating," and LCO 3.8.5, "DC Sources – Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Electrolyte (level and specific gravity) and cell float voltage limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met. Train A batteries are composed of Channel A and Channel C batteries. Train B batteries are composed of Channel B and Channel D batteries.

(continued)

BASES (continued)

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte and cell float voltage are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in the Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A.1, A.2, and A.3

With one or more batteries with one or more battery cell parameters not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because electrolyte level, temperature, cell float voltage, and specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters not within the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing any of the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including float voltage, specific gravity, and electrolyte level of pilot cells.

SR 3.8.6.2

The quarterly inspection of level, specific gravity and float voltage is consistent with IEEE-450 (Ref. 3). In addition, within 7 days of a battery discharge < 105 V or a battery overcharge > 150 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.6.2 (continued)

voltage to drop to ≤ 105 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells (a 10% representative sample of available cells for each battery bank) is > 60°F is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on vendor recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose level, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on vendor recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6.1 (continued)

effects. In addition to this allowance, footnote (a) to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell (low specific gravity cells) or ≥ 2.18 V per cell (AT&T). This value is based on the battery vendor recommendation which states that prolonged operation of cells < 2.13 V (low specific gravity cells) or < 2.18 V (AT&T) can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (low specific gravity cells) or ≥ 1.290 (AT&T) (0.015 [low specific gravity cells] or 0.10 [AT&T] below the vendor fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. Footnote (d) to Table 3.8.6-1 is applicable to Category B float voltage. Footnote (d) requires correction for average electrolyte temperature. The Category B limit specified for specific gravity for each connected cell is ≥ 1.195 (low specific gravity cells) or ≥ 1.280 (AT&T) (0.020 below the vendor fully charged, nominal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6.1 (continued)

specific gravity) with the average of all connected cells >1.205 (low specific gravity cells) or > 1.290 (AT&T) (0.010 below the vendor fully charged, nominal specific gravity). These values are based on vendor's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limit for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limit, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C Allowable Value for float voltage is based on vendor recommendations which state that a cell voltage of 2.07 V or below (low specific gravity cells) or 2.14 V or below (AT&T), under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.195 (low specific gravity cells) or ≥ 1.280 (AT&T) is based on vendor recommendations (0.020 below the vendor recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnotes (b) and (c) to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires specific gravity correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery equalizing recharge. Within 7 days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

3. IEEE-450-1980.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital instrument buses because of the stability and reliability they achieve by being powered from the 125 VDC battery source. The function of the inverter is to provide AC electrical power to the AC vital instrument buses. The AC vital instrument bus can be powered from an AC source via a Class 1E constant voltage regulator or from the inverter connected to the station battery. This configuration provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). There are two inverters per Train (A and B) which totals to four inverters per unit. Specific details on inverters and their operating characteristics are found in the UFSAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital instrument buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst case single failure.

(continued)

BASES (continued)

SAFETY ANALYSIS (continued)

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the systems' instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters (two per train) ensure an uninterruptible supply of AC electrical power to the AC vital instrument buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated AC vital instrument bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverters from a 125 VDC station battery.

This LCO is modified by a Note that allows one inverter to be disconnected from its associated battery for ≤ 24 hours, if the AC vital instrument bus is powered from a Class 1E constant voltage regulator during the period and all other inverters are operable. This allows an equalizing charge to be placed on one battery. If the inverter was not disconnected, the resulting voltage condition might damage the inverter. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital instrument bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

The intent of this Note is to limit the number of inverters that may be disconnected. Only the inverter associated with the single battery undergoing an equalizing charge may be

(continued)

BASES (continued)

LCO (continued)

disconnected. All other inverters must be connected to their associated batteries and aligned to their associated AC vital instrument buses.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are covered in the Bases for LCO 3.8.8, "Inverters – Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital instrument bus becomes inoperable until it is re-energized from its Class 1E constant voltage source regulator.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating," when Condition A is entered with one AC vital instrument bus de-energized. This ensures the AC vital instrument bus is re-energized within 2 hours via the Class 1E constant voltage regulator.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC

(continued)

BASES (continued)

ACTIONS

A.1 (continued)

vital instrument bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital instrument buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital instrument buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital instrument buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR, Chapter 8.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital instrument bus during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

The required inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital instrument buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital instrument bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

The Actions are modified by a Note that identifies required Action A.2.3 is not applicable to the movement of irradiated fuel assemblies in Modes 1 through 4.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains of AC vital instrument buses are required by LCO 3.8.10, "Distribution Systems – Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, operations with a potential for draining the reactor vessel, and operations with a potential for positive

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, inability to immediately suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source regulator.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital instrument buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1 (continued)

for the instrumentation connected to the AC vital instrument buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems – Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and AC vital instrument bus electrical power distribution systems are divided into two trains. Each train has redundant and independent AC, DC, and AC vital instrument bus electrical power distribution subsystems.

The AC primary electrical power distribution system consists of two 4.16 kV Engineered Safety Feature (ESF) buses. Each 4.16 kV ESF bus is normally connected to a offsite source. If the offsite source is de-energized or disconnected, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources – Operating," and the Bases for LCO 3.8.4, "DC Sources – Operating."

The secondary AC electrical power distribution system for each train includes the safety related load centers, and motor control centers shown in Table B 3.8.9-1.

The 120 VAC vital instrument buses are arranged in two channels per subsystem and are normally powered from the inverters. There are four channels designated as A, B, C and D for each unit. The alternate power supply for the vital instrument buses are Class 1E constant voltage source regulators powered from train-related Class 1E motor control centers and its use is governed by LCO 3.8.7, "Inverters – Operating."

There are two independent 125 VDC electrical power distribution subsystems (Train A and Train B). Each subsystem contains two DC power channels. There are four channels designated as A, B, C, and D for each unit.

The list of all required distribution buses is presented in Table B 3.8.9-1. The six electrical power distribution subsystems consist of those components identified by Table B 3.8.9-1. Load breakers not identified by this table do not impact this LCO but may impact supported system LCOs. Load breakers that are required to maintain energized those buses identified by Table B 3.8.9.-1 (e.g. PG to PH) do impact this LCO.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The AC, DC, and AC vital instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC vital instrument bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The six required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC vital instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and AC vital instrument bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Train A and Train B AC, DC, and AC vital instrument bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

(continued)

BASES

LCO (continued)

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, and motor control centers to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE AC vital instrument bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage, or Class 1E constant voltage regulator.

In addition, tie breakers between redundant safety related AC, DC, and AC vital instrument bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are covered in the Bases for LCO 3.8.10, "Distribution Systems – Shutdown."

(continued)

BASES (continued)

ACTIONS

A.1

With one or more required AC buses, load centers, or motor control centers (see Table B 3.8.9.-1), except AC vital instrument buses, in one subsystem inoperable, the remaining AC electrical power distribution subsystem in the other train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers and motor control centers must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one train (PBA or PBB) without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of

(continued)

BASES

ACTIONS

A.1 (continued)

failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With AC vital instrument bus(es) (Channels A or C, or Channels B or D) (see Table B 3.8.9-1) in one train inoperable, the remaining OPERABLE AC vital bus electrical power distribution subsystem is capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital instrument buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter via inverted DC voltage or the Class 1E constant voltage regulator.

Condition B represents one train without adequate AC vital instrument bus power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of OPERABILITY to the remaining vital instrument buses, and restoring power to the affected electrical power distribution subsystem.

(continued)

BASES

ACTIONS

B.1 (continued)

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate AC vital instrument power. Taking exception to LCO 3.0.2 for components without adequate AC vital instrument power, which would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate AC vital instrument power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC vital instrument bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital instrument buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the vital instrument bus distribution system. At this time, an AC train could again become inoperable, and vital instrument bus distribution restored OPERABLE. This could continue indefinitely.

(continued)

BASES

ACTIONS

B.1 (continued)

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With DC bus(es) in one train (see Table B 3.8.9-1) inoperable, the remaining DC electrical power distribution subsystem is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or battery charger.

Condition C represents one train without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining DC buses and restoring power to the affected DC electrical power distribution subsystem.

(continued)

BASES

ACTIONS

C.1 (continued)

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one Condition is entered, and this results in the loss of a required safety function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation.

LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the AC, DC, and AC vital instrument bus electrical power distribution systems are functioning properly, with the required circuit breakers closed and the buses energized. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. Regulatory Guide 1.93, Revision 0, December 1974.

Table B 3.8.9-1
(Units 1, 2, and 3)

TYPE	VOLTAGE	TRAIN A		TRAIN B	
AC safety buses	4160 V 480 V 480 V	ESF Bus PBA-S03 Load Centers PGA-L31, PGA-L33, PGA-L35 Motor Control Centers PHA-M31, PHA-M33, PHA-M35, PHA-M37		ESF Bus PBB-S04 Load Centers PGB-L32, PGB-L34, PGB-L36 Motor Control Center PHB-M32, PHB-M34, PHB-M36, PHB-M38	
DC buses	125 V	CHANNEL A Bus PKA-M41 Distribution Panel PKA-D21	CHANNEL C Bus PKC-M43 Distribution Panel PKC-D23	CHANNEL B Bus PKB-M42 Distribution Panel PKB-D22	CHANNEL D Bus PKD-M44 Distribution Panel PKD-D24
AC vital instrument buses	120 V	CHANNEL A Bus PNA-D25	CHANNEL C Bus PNC-D27	CHANNEL B Bus PNB-D26	CHANNEL D Bus PND-D28

NOTE: Each train of the electrical power distribution system is comprised of the independent AC, DC, and AC vital instrument bus subsystems.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems – Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC vital instrument bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital instrument bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC vital instrument bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies, ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment and components – all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC, DC, and AC vital instrument bus electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and AC vital instrument bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

(continued)

BASES (continued)

ACTIONS

The Actions are modified by a Note that identifies required Action A.2.3 is not applicable to the movement of irradiated fuel assemblies in Modes 1 through 4.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected required features LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, inability to immediately suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC, DC, and AC vital instrument bus electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

(continued)

BASES (continued)

ACTIONS (continued)

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, DC, and AC vital instrument bus electrical power distribution system is functioning properly, with all the required buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal, during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted and the head is slowly removed. The refueling canal is flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

(continued)

BASES

BACKGROUND (continued)

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling canal mix the water to obtain a uniform concentration. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.2, "SHUTDOWN MARGIN—Reactor Trip Breakers Closed."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal to ensure a uniform boron concentration is maintained for reactivity control in the volumes having direct access to the reactor vessel while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) – Reactor Trip Breakers Open," and LCO 3.1.2, "SHUTDOWN MARGIN – Reactor Trip Breakers Closed," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

(continued)

BASES

ACTIONS

A.3 (continued)

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible at greater than or equal to 26 gpm of a solution containing greater than 4000 ppm boron. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 9.1.2.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The Startup Channel Neutron Flux Monitors or Startup Range Monitors (SRMs) and the associated Boron Dilution Alarm System (BDAS) are used during core alterations or movement of irradiated fuel assemblies in containment to monitor the core reactivity condition. The installed SRMs are part of the Excore Nuclear Instrumentation System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

The installed SRMs are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers five decades of neutron flux (1E+5 cps) with a 5% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible indication in the control room and containment. An audible BDAS alarm alerts operators to a possible dilution accident. The excore startup channels are designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE SRMs and the associated BDAS are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

The SRMs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

(continued)

BASES

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 3, 4, and 5, the installed source range detectors and circuitry are required to be OPERABLE by LCO 3.3.12, "Boron Dilution Alarm System (BDAS).

ACTIONS

A.1 and A.2

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, a CHANNEL CHECK minimizes the chance of loss of function due to failure of redundant channels.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational. This SR is an extension of SR 3.3.12 for the Boron Dilution Alarm System CHANNEL CALIBRATION listed here because of its Applicability in these MODES. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

2. UFSAR, Section 15.4.6.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has doors at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment

(continued)

BASES

BACKGROUND (continued)

closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The refueling purge subsystem includes a 42 inch supply penetration and a 42 inch exhaust penetration. The second subsystem, power access purge subsystem, includes an 8 inch supply penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the refueling purge supply and exhaust penetrations are secured in the closed position. The two valves in each of the two power access purge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The refueling purge system is used for this purpose and the valves are closed by the ESFAS in accordance with LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)."

The Power Access Purge System remains operational in MODE 6 and the valves are also closed by the ESFAS.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent.

(continued)

BASES

BACKGROUND (continued)

Equivalent isolation methods must be approved and may include use of devices designed to allow eddy current testing and sludge lancing of the steam generators. Devices which present a substantial restriction to the release of containment atmosphere may be considered equivalent.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Water Level-Fuel Assemblies," LCO 3.9.7, "Refueling Water Level-CEAs," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 3), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge supply and exhaust penetrations. For the OPERABLE containment purge supply and exhaust penetrations, this LCO ensures that these penetrations are isolable by a valve in the Containment Purge Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge valve closure times specified in the UFSAR can be achieved and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.

(continued)

BASES

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge Isolation System not capable of automatic actuation when the purge valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic containment purge isolation signal.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. The CPIAS is tested in accordance with LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)." SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. UFSAR, Section 15.7.4.

3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation–High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Essential Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to a resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the SDC System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing of the SDC pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the SDC pump does not result in a challenge to the fission product barrier. SDC and Coolant Circulation–High Water Level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop includes an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level - Fuel Assemblies."

(continued)

BASES

APPLICABILITY (continued)

Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation–Low Water Level."

ACTIONS

SDC loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading an irradiated fuel assembly, is a prudent action under this condition.

A.3

If SDC loop requirements are not met, actions shall be initiated and continued in order to satisfy SDC loop requirements.

(continued)

BASES

ACTIONS (continued)

A.4

If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the containment building. The 4 hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.

REFERENCES

1. UFSAR, Section 5.4.7.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation–Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Essential Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation–Low Water Level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

Both SDC pumps may be aligned to the Refueling Water Tank (RWT) to support filling the refueling cavity or for performance of required testing.

The LCO is modified by a Note that allows a required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System. MODE 6 requirements, with a water level ≥ 23 ft above the reactor vessel flange, are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level."

(continued)

BASES

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 23 ft above the reactor vessel flange, the Applicability will change to that of LCO 3.9.4, "Shutdown Cooling and Coolant Circulation–High Water Level," and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3

If no SDC loop is in operation or no SDC loops are OPERABLE, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2

Verification that the required pump that is not in operation is OPERABLE ensures that an additional SDC 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 the required pump. 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.

REFERENCES

1. UFSAR, Section 5.4.7.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level-Fuel Assemblies

BASES

BACKGROUND

The movement of fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated. During refueling this maintains sufficient water level in the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 33% of 10 CFR 100 limits, which meets the intent of the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated in the reactor vessel are irradiated. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, and HVAC operations which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Water Level - CEAs

BASES

BACKGROUND

The movement of CEAs within the reactor vessel, when irradiated fuel assemblies are seated in the reactor vessel requires a minimum water level of 23 ft above the top of the irradiated fuel. During refueling this maintains sufficient water level in the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 33% of 10 CFR 100 limits, which meets the intent of the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of CEA's the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

A minimum refueling water level of 23 ft above irradiated assemblies seated within the reactor vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.7 is applicable during movement of CEAs within the reactor vessel when irradiated fuel assemblies are seated within the reactor vessel. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel, all operations involving movement of CEAs within the reactor vessel shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of movement of CEAs shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions and HVAC operations, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.