ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA US) [Ronda.Pederson@areva.com]
Sent:	Friday, December 12, 2008 4:39 PM
To:	Getachew Tesfaye
Cc:	PORTER Thomas (EXT); DUNCAN Leslie E (AREVA US); BENNETT Kathy A (OFR) (AREVA US); DELANO Karen V (AREVA US)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 101 (931), FSARCh. 16
Attachments:	RAI 101 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 101 Response US EPR DC.pdf" provides technically correct and complete responses to 54 of the 71 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 101 Questions 16-58, 16-59, 16-60, 16-63, 16-64, 16-65, 16-71, 16-73, 16-75, 16-78, 16-80, 16-81, 16-83, 16-84, 16-85, 16-86, 16-87, 16-89, 16-91, 16-92, 16-93, 16-94, 16-95, 16-96, 16-97, 16-98, 16-99, 16-100, 16-101, 16-102, 16-104, 16-107, 16-109, 16-110, 16-112, 16-113, 16-114, 16-115, 16-116, 16-117, 16-118, 16-122, 16-123, and 16-124.

The following table indicates the respective page(s) in the response document, "RAI 101 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 101 — 16-54	2	2
RAI 101 — 16-55	3	3
RAI 101 — 16-56	4	4
RAI 101 — 16-57	5	5
RAI 101 — 16-58	6	6
RAI 101 — 16-59	7	7
RAI 101 — 16-60	8	8
RAI 101 — 16-61	9	9
RAI 101 — 16-62	10	10
RAI 101 — 16-63	11	11
RAI 101 — 16-64	12	12
RAI 101 — 16-65	13	13
RAI 101 — 16-66	14	14
RAI 101 — 16-67	15	15
RAI 101 — 16-68	16	16
RAI 101 — 16-69	17	17
RAI 101 — 16-70	18	18
RAI 101 — 16-71	19	19
RAI 101 — 16-72	20	20
RAI 101 — 16-73	21	22
RAI 101 — 16-74	23	23
RAI 101 — 16-75	24	24
RAI 101 — 16-76	25	25
RAI 101 — 16-77	26	26
RAI 101 — 16-78	27	27
RAI 101 — 16-79	28	28

RAI 101 — 16-80	29	29
RAI 101 — 16-81	30	30
RAI 101 — 16-82	31	31
RAI 101 — 16-83	32	32
RAI 101 — 16-84	33	33
RAI 101 — 16-85	34	34
RAI 101 — 16-86	35	35
RAI 101 — 16-87	36	36
RAI 101 — 16-88	37	37
RAI 101 — 16-89	38	38
RAI 101 — 16-90	39	39
RAI 101 — 16-91	40	40
RAI 101 — 16-92	41	41
RAI 101 — 16-93	42	42
RAI 101 — 16-94	43	43
RAI 101 — 16-95	44	44
RAI 101 — 16-96	45	45
RAI 101 — 16-97	46	46
RAI 101 — 16-98	47	47
RAI 101 — 16-99	48	48
RAI 101 — 16-100	49	49
RAI 101 — 16-101	50	50
RAI 101 — 16-102	51	54
RAI 101 — 16-103	55	55
RAI 101 — 16-104	56	56
RAI 101 — 16-105	57	57
RAI 101 — 16-106	58	58
RAI 101 — 16-107	59	59
RAI 101 — 16-108	60	60
RAI 101 — 16-109	61	61
RAI 101 — 16-110	62	62
RAI 101 — 16-111	63	63
RAI 101 — 16-112	64	64
RAI 101 — 16-113	65	66
RAI 101 — 16-114	67	67
RAI 101 — 16-115	68	68
RAI 101 — 16-116	69	69
RAI 101 — 16-117	70	70
RAI 101 — 16-118	71	71
RAI 101 — 16-119	72	72
RAI 101 — 16-120	73	73
RAI 101 — 16-121	74	74
RAI 101 — 16-122	75	75
RAI 101 — 16-123	76	76
RAI 101 — 16-124	77	77

A complete answer is not provided for 17 of the 71 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 101 — 16-54	February 19, 2009

RAI 101 — 16-55	February 19, 2009
RAI 101 — 16-56	February 19, 2009
RAI 101 — 16-57	February 19, 2009
RAI 101 — 16-61	February 19, 2009
RAI 101 — 16-66	February 19, 2009
RAI 101 — 16-67	February 19, 2009
RAI 101 — 16-68	February 19, 2009
RAI 101 — 16-69	February 19, 2009
RAI 101 — 16-70	February 19, 2009
RAI 101 — 16-72	February 19, 2009
RAI 101 — 16-74	February 19, 2009
RAI 101 — 16-76	February 19, 2009
RAI 101 — 16-77	February 19, 2009
RAI 101 — 16-79	February 19, 2009
RAI 101 — 16-82	February 19, 2009
RAI 101 — 16-103	February 19, 2009

Sincerely,

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Wednesday, November 12, 2008 8:14 PM
To: ZZ-DL-A-USEPR-DL
Cc: Hien Le; Michael Marshall; Peter Hearn; Joseph Colaccino; John Rycyna
Subject: U.S. EPR Design Certification Application RAI No. 101 (931), FSARCh. 16

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 16, 2008, and discussed with your staff on October 28, 2008. None of the draft RAI question were modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
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Hearing Identifier: AREVA_EPR_DC_RAIs
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Response to

Request for Additional Information No. 101 (931), Revision 0

11/12/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: TS Section 3.4

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-54:

FSAR subsection 15.0.0.3.1, Design Plant Conditions and Initial Conditions, states, in part, "A thermal design flow of 119, 692 gpm per loop is used in the accident analysis for the RCS response. This thermal design flow is the minimum allowed by plant TSs. The analyses bound up to five percent SG tube plugging." In addition, the Westinghouse STS, NUREG-1431, specifies both the COLR limit and the SG tube plugging limit in LCO 3.4.1.

Explain the accounting for the the minimum flow rate of 119,692 gpm per loop, as listed in FSAR Table 15.0-5, in LCO 3.4.1. Based on the accounting, revise LCO 3.4.1 and related information in the TS bases B 3.4.1, as appropriate.

Response to Question 16-54:

A response to this Question will be provided by February 19, 2009.

Question 16-55:

Revise LCO 3.4.11, SR 3.4.11.4 and the associated bases B 3.4.11 to state LCO 3.4.11.c as a Note.

LCO line items are for listing of the minimum number of plant equipment to be OPERABLE. Special operating requirement such as "Reactor coolant pumps shall not be started unless the secondary side water temperature of each steam generator is less than or equal to 50 degree F above each of the RCS cold leg temperature" should be listed as a Note to be consistent with the format used in the STS.

Response to Question 16-55:

A response to this Question will be provided by February 19, 2009.

Question 16-56:

Provide information to justify the addition of Condition A and its associated Required Actions and Completion Times. Revise TS 3.4.4 and related information in the TS Bases B 3.4.4, as appropriate.

Condition A allows 3-Loop operation for up to 2 hours and entails a restart of the 4th Reactor Coolant Pump (RCP). In the Westinghouse STS, NUREG-1431, the same condition would require a unit shutdown to Mode 3. Moreover, provisions are needed to ensure that the cause of the problem is identified and corrected before restarting the idle loop or this LCO could possibly put the plant in an unanalyzed condition.

This information will be used to ensure all of the conditions specified are compatible with the actions to restore LCO operability or to exit the proposed LCO applicability.

Response to Question 16-56:

A response to this Question will be provided by February 19, 2009.

Question 16-57:

Confirm that a minimum flow rate of 2200 gpm through the core is required for the performance of SR 3.4.6.1. Revise SR 3.4.6.1 and the related information in the bases B 3.4.6, as appropriate.

The BACKGROUND of TS Bases B 3.4.6 states: "In MODE 4, the flow provided by one RCP or two RHR loops is adequate for decay heat removal." Moreover, EPR FSAR Tier 2, Table 6.3-2 lists a normal flow rate of 2200 gpm for each LHSI/RHR pump (4400 gpm for 2 pumps running).

Response to Question 16-57:

A response to this Question will be provided by February 19, 2009.

Question 16-58:

Revise the second paragraph of TS Bases B 3.6.4, APPLICABLE SAFETY ANALYSES, to add discussion of results from the worst case MSLB for the maximum peak containment internal pressure.

The first paragraph of the same section in TS B 3.6.4, states "The worst case MSLB generates larger mass and energy release than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure standpoint." The results from the worst case MSLB should be discussed when MSLB is determined as the limiting event.

Response to Question 16-58:

The same initial pressure was used for the loss of coolant accident (LOCA) and main steam line break (MSLB) analysis. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases 3.6.4, “Applicable Safety Analysis” will be revised to reflect that the MSLB is the limiting event. Note that P_a was conservatively chosen to bound the peak MSLB pressure.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.4 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-59:

TS Bases B 3.6.4, Containment Pressure.

Revise the TS bases B 3.6.4 to indicate where Reference 2, "10 CFR 50, Appendix K, ECCS Evaluation Models" is used within the body of the bases. Delete it from REFERENCES if not applicable.

Response to Question 16-59:

Reference 2 will be deleted from the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.6.4 References.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.4 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-60:

Provide justification for the minimum run time of 15 minutes with heaters operating in SR 3.6.7.1. Revise TS 3.6.7, as appropriate.

The specified minimum 15-minute run time is not consistent the discussion in the TS bases for 3.6.7.1. Further more, WOG STS requirements for a similar application SR 3.6.13.1 states "operate each SBACS train for [>or= 10 continuous hours with heaters operating or (for systems without heaters) >or= 15 minutes]."

This RAI applies also to SR 3.7.10.1 in TS 3.7.10, and SR 3.7.12.3 in TS 3.7.12.

Response to Question 16-60:

The 15-minute run time is consistent with the guidance provided by RG 1.52, Revision 3, Section 6.1. U.S. EPR FSAR Tier 2, Chapter 1, Table 1.9-2 references RG 1.52, Revision 3 for the U.S. EPR ventilation systems. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.6.7, 3.7.10, and 3.7.12 will be revised to clarify the use of RG 1.52, Revision 3.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.6.7, 3.7.10, and 3.7.12 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-61:

Revise TS 3.6.8 and related information in the TS Bases 3.6.8 to add a surveillance requirement (SR) to test a sample of trisodium phosphate (TSP) to ensure the solubility and buffering ability of TSP after exposure to the containment environment.

Response to Question 16-61:

A response to this Question will be provided by February 19, 2009.

Question 16-62:

Provide additional information to justify the tolerance of 3% for the setpoint setting of MSSVs in SR 3.7.1.1.

ASME Code, Section III, NC 7000 (Subsection NC 7512) is listed as Reference 2 in the TS bases B 3.7.1. ASME Code Subsection NC 7512 states, in part, "the set pressure tolerance plus or minus shall not exceed the following: 2 psi (15 kPa) for pressures up to and including 70 psi (500 kPa), 3% for pressures over 70 psi (500 kPa) up to and including 300 psi (2000 kPa), 10 psi (70 kPa) for pressures over 300 psi (2000 kPa) up to and including 1000 psi (7000 kPa), and 1% for pressures over 1000 psi (7000 kPa). The set pressure tolerance shall apply unless a greater tolerance is established as permissible in the Overpressure Protection Report (NC-7200)."

Response to Question 16-62:

As noted in the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for SR 3.7.1.1:

> "The SR allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure."

The ASME Code requirement is met provided the ± 1% tolerance is met during preservice and inservice testing. Main steam safety valves (MSSV) may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-63:

Revise TS 3.7.10 to correctly incorporate the changes as approved under TSTF-448, Revision 3.

The approved changes under TSTF-448 to the descriptions of Conditions A, B and E should be incorporated verbatim to reflect the intended purpose and to be consistent with conforming changes made in the TS bases B 3.7.10.

Response to Question 16-63:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.7.10 and Bases will be revised to conform to TSTF-448, Revision 3.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.10 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-64:

TS Bases B 3.4.6, RCS Loops-MODE 4.

Revise the description for Actions A.1 and A.2 in the TS Bases B 3.4.6 to add a discussion for the selected Completion Time of 24 hours for Action A.2.

Justification for the 24-hour CT was not provided in the associated TS bases.

This is needed to ensure completeness of supporting information in the TS bases.

Response to Question 16-64:

Text will be added to the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 Bases for Action A.2. The 24 hour Completion Time also will be provided.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-65:

Provide justification for not providing TS requirements for Decay Time prior to fuel handling. Revise TS 3.9 and related information in the bases, as appropriate.

In EPR FSAR Subsection 15.0.3.10, Fuel Handling Accident, a minimum Decay Time of 34 hours is assumed as an initial condition in the accident analysis. Also, 10CFR50.36(c)(2)(ii) requires a limiting condition for operation (LCO) to be established for "a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis."

Response to Question 16-65:

A new Technical Specification and Bases will be added for a minimum decay time requirement.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed markup.

Question 16-66:

TS Bases B 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits.

Provide further explanation for measurement error in the second paragraph of the LCO 3.4.1 Bases discussion of LCO.

The Bases state "RCS total flow contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators." The explanation should explicitly identify and describe the source of the measurement error. For example, the STS Bases provide a discussion of this measurement error due to fouling of the feedwater venturi used in the operating plants. A comparable discussion was not provided in the EPR TS Bases.

This information will be used to ensure the completeness of information provided in the TS Bases.

Response to Question 16-66:

A response to this Question will be provided by February 19, 2009.

Question 16-67:

TS Bases B 3.4.10, Pressurizer Safety Valves.

Clarify the statement "the overpressure protection analysis (Ref. 3) is also based on operation of three PSRVs" in the discussion of Applicable Safety Analyses in the TS Bases B 3.4.10.

ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants, 2004 is listed as Reference 3 in the EPR TS bases. A Westinghouse topical report (WCAP-7769) on overpressure protection is also listed as Reference 3 in the STS Bases 3.4.10.

This information will be used to ensure consistency between the EPR TS and the referenced source document.

Response to Question 16-67:

A response to this Question will be provided by February 19, 2009.

Question 16-68:

TS Bases B 3.4.11, Low Temperature Overpressure Protection (LTOP) System.

Justify that LCO 3.4.11, items a, b, c, and d, ensures LTOP is Operable and assure that it was developed from FSAR 5.2.2.2.2. Several of the mass and heat input transients were identified as being covered by other analyses without further discussion. Revise TS Bases B 3.4.11 to provide proper references or include any additional information needed to justify that LCO 3.4.11, items a, b, c, and d, ensures LTOP is Operable.

This additional information will be used to ensure that the LCO statement is accurate and complete, and adequately justified.

Response to Question 16-68:

A response to this Question will be provided by February 19, 2009.

Question 16-69:

TS 3.4.11, Low Temperature Overpressure Protection (LTOP) System.

Provide the consequences if LCO 3.4.11 Condition A cannot be met. There appears to be no specific condition that would require entry into the related LCO under the circumstances given.

This additional information will be used to ensure that the LCO statement is adequately justified or needed.

Response to Question 16-69:

A response to this Question will be provided by February 19, 2009.

Question 16-70:

TS Bases B 3.4.11, Low Temperature Overpressure Protection (LTOP) System.

Provide a description of the most probable vent flowpath that will be opened to achieve the related requirement associated with LCO 3.4.11. Include this information in the Bases to LCO 3.4.11.

This additional information will be used to ensure that the LCO statement is complete and adequately justified.

Response to Question 16-70:

A response to this Question will be provided by February 19, 2009.

Question 16-71:

TS Bases B 3.4.12, RCS Pressure Isolation Valve (PIV) Leakage.

Provide the necessary changes to make SR 3.4.13.1 and the Bases consistent.

SR 3.4.13.1, Frequency, currently states "in accordance with the Inservice Testing Program" and the Bases discussion for SR 3.4.13.1 identifies a frequency of 24 months. Although they are the similar in value, the requirements of the more restrictive frequency should be used; therefore, the TS and Bases should be expressed in identical units, as applicable.

This requested change in information will be used to ensure consistent use information and terminology for clarity purposes.

Response to Question 16-71:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, SR 3.4.13.1 Frequency will be revised to be consistent with the STS by adding "24 months".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.13 will be revised as described in the response and indicated on the enclosed markup.

Question 16-72:

TS 3.4.14, RCS Leakage Detection Instrumentation

Provide a technical justification for the absence of CHANNEL (DIVISION) OPERATIONAL TEST, performed on the containment atmosphere radioactivity monitor at a frequency of every 92 days.

This additional information will be used to ensure that surveillance requirements are complete.

Response to Question 16-72:

A response to this Question will be provided by February 19, 2009.

Question 16-73:

EDITORIAL

1. In the Bases References, there is a document that is only identified as "Chapter 5 or 15". Provide a more specific title and specific "15.x.x.x" subsection identity. This occurs in LCOs 3.4.1, 3.4.2, 3.4.4, 3.4.5, 3.4.9, through 3.4.14.
2. In LCO 3.4.2, Tavg (subscript) is used. In Condition A and SR 3.4.2.1, Tavg is used rather than Tavg (subscript). Make these terms consistent.
3. In Bases of LCO 3.4.3, correct the spelling of BACKGROUND in the left hand margin header title. The two letters, "K" and "C", are transposed for "BAKCGROUND".
4. In the Bases B 3.4.5, define "CRDM" in Applicable Safety Analyses, the last sentence of the first paragraph to read, "Such a transient" mechanical failure of a Control Rod Drive Mechanism (CRDM).
5. In the Bases discussion of Applicable Safety Analyses for LCO 3.4.5, replace the word "irregardless" with the word "regardless", in the third sentence of the first paragraph. The word "irregardless" is an informal non-standard word.
6. In LCO 3.4.6, correct the format error for the placement of the Note to Required Action A.2. It should be indented and written directly in-line after "A.2".
7. In the Applicable Safety Analyses Bases for LCO 3.4.9, the second paragraph needs to identify from what reference (FSAR?) Chapter 15.x.x.x is quoted.
8. In the Bases Background for 3.4.10 second paragraph, the staff recommends adding "(2788 psig)" after "2803 psia" for ease of use by operators who used to reading pressure in gage value from plant instruments.
9. In the Bases Background for LCO 3.4.10, change the system name to Reactor Protection System (RPS) in the first sentence to read, "The Pressurizer Safety Relief Valves (PSRVs), in conjunction with the Reactor Protection System (RPS)." Contrary to this entry, LCO and Bases consistently refer to the system as Protection System (PS).
10. In the LCO 3.4.10, Applicability Bases discussion, 2nd paragraph, 1st sentence, there is a misplaced period at the end of PTLR that must be removed for clarity.
11. In the Bases for Action A.1 in LCO 3.4.10, add the word "relief" to the third sentence to read "An inoperable safety relief valve". Contrary to this entry, LCO and Bases consistently refer to the component as "safety relief valve".
12. In the Bases for Applicable Safety Analyses in LCO 3.4.12, punctuate the first sentence in the second paragraph correctly by adding a closing parenthesis.
13. In LCO 3.4.13, properly align (indent) the logical connector "OR" between the Required Actions A.2.1 and A2.2.
14. In the LCO 3.4.15, Bases for Actions B.1, last sentence of second paragraph: The hyphen should be removed between "transient" and "specific" and the comma needs to be removed between "to" and "power".
15. In LCO 3.4.16, underline the logical connector AND between Required Action B.1 and Required Action B.2.

16. In the Bases for SR 3.4.16.1, the first sentence in the last paragraph refers to the Frequency as being in SR 3.4.17.1 when the reference should be SR 3.4.16.1.

Response to Question 16-73:

1) References to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, chapters and sections were reviewed. In some cases, a general reference to a chapter is appropriate and consistent with the STS and operating plant Technical Specifications. For those cases where a more detailed reference is appropriate, the References sections of the Bases will be revised.
2) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.4.2 will be revised.
3) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for LCO 3.4.3 will be revised.
4) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.5, Applicable Safety Analysis, will be revised.
5) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.5, Applicable Safety Analysis, will be revised.
6) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.4.6 will be revised.
7) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.9, Applicable Safety Analysis, will be revised.
8) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.10, Background, will be revised.
9) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.10, Background, will be revised.
10) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.10, Applicability, will be revised.
11) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.10, Actions, will be revised.
12) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.12, Applicable Safety Analysis, will be revised.
13) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.4.13 will be revised.
14) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.15, Actions, will be revised.
15) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.4.16 will be revised.
16) U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for SR 3.4.16.1 will be revised.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed markup.

Question 16-74:

TS 3.4.17, RCS Loops - Test Exceptions.

Provide a technical justification as to why there are no provisions for the use of a redundant interlock during applicable Special Test Exception to prevent THERMAL POWER from exceeding the limit. Confirm that the EPR design has a Low Power Reactor Trip permissive that enables an additional degree of core protection during this test. If so, provide additional justification for not testing this function to ensure the Operability of this protective function prior to entering this LCO.

This additional information will be used to ensure that operation limits, surveillance requirements, and restrictions are adequate and complete for the proposed LCO.

Response to Question 16-74:

A response to this Question will be provided by February 19, 2009.

Question 16-75:

TS 3.4.6, RCS Loops - MODE 4.

Provide further clarification for Required Action B.2 that states "initiate action to restore one loop to OPERABLE status and operation". Revise TS 3.4.6 and the asociated Bases, as appropriate.

LCO 3.4.6 specifies two distict plant equipment configuration: either (1) Two RCS loops shall be OPERABLE and one RCS loop shall be in operation, or (2) Three RHR loops shall be OPERABLE and two RHR loops shall be in operation. Condition B describes two scenarios: either (1) Two or more required loops inoperable, or (2) Required loop(s) not in operation. In order to meet the LCO requirements imposed on the RHR configuration, the Required Action B.2 should restore the two required RHR loops (not just one loop) to OPERABLE status and operation.

This information is needed to ensure adequacy of the proposed Required Action to resolve the identified Condition.

Response to Question 16-75:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 and Bases will be revised to restore two required residual heat removal (RHR) loops to OPERABLE status and operation.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-76:

TS 3.4.6, RCS Loops - MODE 4.

Provide the methodology for determining the SG secondary side water level "20%" limit that is required in SR 3.4.6.3 and verify the location of this limit in the FSAR.

The information will be used to ensure that all TS specific operating parameters are verified as correct based upon values stated in the FSAR.

This RAI applies also to TS 3.4.7.

Response to Question 16-76:

A response to this Question will be provided by February 19, 2009.

Question 16-77:

TS 3.4.6, RCS Loops - MODE 4.

Justify the exclusion of restricting a reactor coolant pump startup with any RCS cold leg temperature less than the Low Temperature Overpressure Protection (LTOP) arming temperature. Also justify the exclusion of the secondary temperature differential restrictions when RCPs and LHSI pumps are removed from operation for a limited duration in LCO 3.4.6 .

The information will be used to ensure that all operating requirements and restrictions are adequately defined in the proposed LCOs.

Response to Question 16-77:

A response to this Question will be provided by February 19, 2009.

Question 16-78:

TS 3.4.7, RCS Loops - MODE 5, Loops Filled.

Provide a clarification and revisions, as necessary for consistency, as to the proper designation is for the "LHSI/RHR" pump and related components in the RHR loop will be for LCOs 3.4.6, 3.4.7 and 3.4.8 in this GTS. In the "Note" for LCO 3.4.6, there is a reference to an LHSI pump and in the Bases for SR 3.4.6.3, there is a reference to an "LHSI/RHR" pump, which is correct. In the "Note" for LCO 3.4.7 and in SR 3.4.7.3, there is a reference to an LHSI pump. The 1st and 2nd paragraphs of the Bases, Background section, identifies a "RHR pump, heat exchanger, etc." as does the Bases for SR 3.4.7.3. The same inconsistency exists in SR 3.4.8.2 and the supporting Bases, Background and LCO sections.

This additional information will be used to ensure no ambiguity is created by inconsistent in terminology.

Response to Question 16-78:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.4.6, 3.4.7 and 3.4.8 will be revised to reference an “LHSI pump” and an “LHSI heat exchanger”, where applicable.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.4.6, 3.4.7 and 3.4.8 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-79:

TS Bases B 3.4.9, Pressurizer.

Provide a clarification for the arrangement of the four heater groups and the half-capacity groups as presented in the Bases, LCO discussion. It is not clear that the half capacity heater groups provide the required capacity is met by the half capacity groups.

Discussion: If there are 12 heaters divided into four groups, then there are three heaters in each group. If each heater is rated at 24 kW, then each group is rated at 72 kW. The LCO states each group has a capacity of >144kW. Therefore, each "half-group" must be equal to a "full group" to be equivalent to 144kW (72 + 72). If it takes two groups to maintain the RCS at normal operating pressure, then the LCO requirement of 3 heater groups Operable to maintain the RCS Operable represents only a verified 50% capacity over the minimum required capacity.

Justify not requiring the four heater groups being Operable. In the Bases, furnish the design value number for pressurizer heaters currently designated Q kW and the half-value currently designated Q/2 kW in LCO, the second paragraph. It appears the pressurizer heater design value is currently designated with a place holder. Provide the location in the FSAR for the above discussion. Also, identify the location in the FSAR for the statement that the pressurizer water volume of 1240 cubic feet is equivalent to a 75% water level.

The Staff will use this change to ensure that the LCO statement is adequately justified in the supporting Bases and then verified as correct per the FSAR.

Response to Question 16-79:

A response to this Question will be provided by February 19, 2009.

Question 16-80:

TS Bases B 3.4.6, RCS Loops - MODE 4.

Provide a discussion for Required Action A.2 and its associated Completion Time in the TS Bases.

The proposed Required Actions and the selected Completion Times for an identified TS Condition should be supported by a discussion in the TS Bases, using information from the FSAR. The STS bases can be used as an example to determine the right level of content and detail that needs to be included in the bases.

This is needed to ensure complete supporting information is provided in the TS Bases.

Response to Question 16-80:

A separate Required Action A.2 will be provided in the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 Bases.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-81:

TS 3.4.15, RCS Specific Activity.

Justify EPR GTS, Section 3.4.15, not fully reflecting the implementation of TSTF-490, Revision 1; or revise EPR GTS Section 3.4.15 to fully reflect proper implementation of TSTF -490, if determined to be applicable.

The bases indicate that TSTF-490 is incorporated into EPR GTS, but it does not seem to be fully implemented.

This additional information will be used to ensure that the applicable LCO correctly considered TSTF -490, as appropriate.

Response to Question 16-81:

TSTF-490-A, Rev. 0 is the current approved version.

Changes were made to the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for 3.4.15 to include vendor specific information for the U.S. EPR. The Applicable Safety Analyses (ASA) discussions in the TSTF-490-A, Rev. 0 Traveler are different for each vendor. U.S. EPR specific information was provided in the ASA discussion in the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.15 Bases. Additional changes were incorporated that were not design related. These changes will be revised to conform the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.15 Bases to TSTF-490-A, Rev. 0.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.15 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-82:

TS 3.4.5, RCS Loops - MODE 3.

Provide a the technical justification for the less restrictive change to the Completion Time from one to two hours for Required Action C.1 with regards to the capabilities of the Control Rod Drive Control System (CRDCS). Typically, a one hour response time is considered appropriate when the plant is placed in an unanalyzed condition.

This justification is being requested to ensure the accuracy and completeness of the LCO

Response to Question 16-82:

A response to this Question will be provided by February 19, 2009.

Question 16-83:

TS Bases B 3.9.3, Containment Penetrations.

Provide the design details of the hatch bolts capable of supporting the hatch dead weight and its associated loads, together with a sketch of the hatch showing bolts' locations and specifications.

EPR TS bases 3.9.3's BACKGROUND states, in part, "The containment equipment hatches, part of the containment pressure boundary, provide access for moving large equipment into and out of the containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch must be closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced." Details of the bolt design are needed to support LCO 3.9.3.a requirements regarding the number of hatch bolts required to ensure the installed configuration meets the intended safety function. STS has bracketed the four-bolt configuration as preliminary information requiring further details on the hatch weight, bolt material and size to be provided by an applicant who wants to adopt this option. These brackets were removed in the EPR TS.

Response to Question 16-83:

U.S. EPR FSAR Tier 2, Section 3.9.3 has been deleted and replaced with a Limited Condition for Operation (LCO) that provides the requirements for a delay time for irradiated fuel assemblies. The Decay Time LCO and Bases do not discuss the containment equipment hatch. Two other LCO sections discuss the containment equipment hatch. U.S. EPR FSAR Tier 2, LCO 3.9.4 Required Action A.4 and LCO 3.9.5 Required Action B.3 will be modified to delete the reference to the four bolts and refer to the containment equipment hatch as being secure.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.9.4 and 3.9.5 will be revised as described in the response and indicated on the enclosed markup.

Question 16-84:

TS Bases B 3.5.1, Accumulators.

Provide a technical justification or revision to ensure that this LCO is consistent with the information in the applicable bases.

The Bases, Applicability section, last portion of 3rd paragraph, defines the role for accumulator trains #3 & #4 where these subsystems need to be Operable during RCS cool down operations. Therefore, determine if this LCO needs to be revised or a second LCO needs to be added to ensure Operability for two required accumulators. In addition, provide a discussion or an appropriate revision that will explain the affect of maintaining these systems Operable on the "Note" applicable to SR 3.5.1.5.

The information is required to ensure the completeness and accuracy of this LCO and SR.

Response to Question 16-84:

Reconnection of an accumulator to the reactor coolant system (RCS) during cooldown is an operational consideration. Discussion of this evolution was included in the Bases as clarification. Without this explanation, the Bases would imply that all four accumulator isolation valves are closed with RCS pressure below 1000 psig. Any of the accumulators can be de-pressurized as described in the revised Applicability discussion of U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases. The train 3 or 4 accumulators are preferred, but are not a requirement. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases Applicability discussion will be revised to delete the reference to trains 3 and 4. Cooldown of the RCS can be accomplished without reconnecting an accumulator and one is not required for safety. Operability of accumulators for this evolution is not needed.

The "Note" that precedes U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.5.1.5 does not require revision because the accumulator reconnection takes place at approximately 400 psig, well below the Applicability for this LCO.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-85:

TS Bases B 3.5.2, ECCS - Operating.

Provide additional information and justification for the following statement in the Bases, Applicability section, second paragraph, second sentence, that states "below 356oF, the Protection System signal setpoint is manually bypassed by operator, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS-Shutdown." Revise the Bases accordingly to reflect this additional information.

The information is required to ensure the completeness and accuracy of this LCO.

Response to Question 16-85:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases will be revised to include additional information.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-86:

TS 3.7.1, Main Steam Safety Valves (MSSVs)

Explain why a Required Action similar to Required Action A.1 is not necessary for Condition B.

Required Action A.1 states "[v]erify associated Main Steam Relief Train is OPERABLE." The verification is required since MSSVs together with the MSRT are credited in the accident analyses. In the TS Bases for LCO 3.7.1, the discussion for Action B.1 states "with two MSSVs inoperable on the same SG, the resulting relief capacity of the affected SG is 50% (taking into account the MSRT) of the full load steam generation per SG." Since MSRT availability is also required for Condition B, a similar verification required to be performed.

This information is needed to ensure adequacy of all proposed Required Actions.

Response to Question 16-86:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.1 and Bases will be revised to allow one main steam safety valve to be out of service. Condition B will be deleted. U.S. EPR FSAR Tier 2, Section 10.3.2.2 will be revised to conform to the change to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.1 Bases.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 10.3.2.2 and Chapter 16, Technical Specifications Section 3.7.1 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-87:

TS Bases B 3.7.12, Safeguard Building Controlled Area Ventilation System

Clarify the statement in the TS Bases B 3.7.12 for Safeguard Building Controlled Area Ventilation System (SBVS), "there is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SBVS" in the discussion of SRs 3.7.12.6 and 3.7.12.7.

This statement appears to be copied from the content of TS Bases 3.6.6 for the Shield Building. In that case, a separate LCO refers to TS 3.6.7 for the Annulus Ventilation System (AVS). TS 3.7.12 covers both the building boundary and the operation of SBVS safety-related equipment.

Response to Question 16-87:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Surveillance Requirements 3.7.12.6 and 3.7.12.7 Bases will be revised to delete the reference to other LCO and surveillance requirements.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.12 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-88:

TS Bases B 3.7.7, Component Cooling Water (CCW) System

Provide additional information to resolve an inconsistency in the TS bases B 3.7.7.

The Background second paragraph states "each safety related train includes a full capacity pump," while the Applicable Safety Analyses first paragraph states "the design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the In-containment Refueling Water Storage Tank (IRWST) by cooling the Low Head Safety Injection System heat exchanger."

This information is required to ensure the information in the bases accurately reflect the system design described in the EPR FSAR.

Response to Question 16-88:

As discussed in U.S. EPR FSAR Tier 2, Chapter 9, Section 9.2.2.2, the component cooling water system (CCWS) utilizes one pump per train to supply the associated train related functions, independent of the operation of other CCWS trains and their pumps. For accident mitigation, two of four CCWS trains are required to be available for required heat removal function.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-89:

TS Bases B 3.7.8, Essential Service Water (ESW) System.

Add FSAR Section 9.2.5 as Reference 5. Also, on Page B 3.7.8-2, in the 4th paragraph, change from "Reference 1" to "Reference 5".

The EPR TS 3.7.8 combines requirements from both WOG STS 3.7.8, Service Water System, and STS 3.7.9, Ultimate Heat Sink. Applicable references used to support the description of the Ultimate Heat Sink need to be included in the EPR TS bases B 3.7.8.

This is needed to ensure information provided in the TS Bases is accurate and complete.

Response to Question 16-89:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.8 Bases will be revised to include the applicable references.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.8 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-90:

TS Bases B 3.7.1, Main Steam Safety Valves (MSSVs).

Explain the inconsistency between the TS bases B 3.7.1, FSAR Table 15.2-1 and requirements of ASME Code, Section III, Article NC-7000.

ASME Code, Section III, Article NC-7000, Class 2 Components is listed as Reference 1 regarding lifting setpoints for the MSSVs. Article NC-7512.1, Antichattering and Lift Requirements, states "Safety valves shall be constructed to operate without chattering and to attain rated lift at a pressure which does not exceed the set pressure by more than 3% or 2 psi (15 kPa), whichever is greater." The lifting setpoints for MSSVs in EPR TS are consistent with information provided in FSAR Table 15.2-1, however, FSAR Table 15.2-1 shows each MSSV delivers its rated lift capacity immediately at the onset of the MSSV opening at the setpoint pressure. Justify assuming that the MSSV delivers its rated lift capacity immediately at the onset of the MSSV opening at the setpoint pressure.

This information is required to ensure the accident analysis assumptions reflect the as-procured equipment performance.

Response to Question 16-90:

The flow area of the main steam safety valves (MSSV) is calculated based on rated flow rate being achieved when system pressure is equal to nominal setpoint pressure, plus 3% accumulation. The MSSV setpoint pressure used in the safety analysis evaluations is calculated as the nominal setpoint pressure, plus 3% setpoint uncertainty (to account for operational drift). Since accumulation and uncertainty have the same value of 3%, rated flow will be achieved at initial valve opening. The MSSV nominal setpoint values and uncertainties assumed in the safety analyses are provided in U.S. EPR FSAR Tier 2, Chapter 15, Table 15.0-9.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-91:

TS Bases B 3.7.1, Main Steam Safety Valves (MSSVs).

Revise the TS bases B 3.7.1 to indicate the location in the body of the Bases of Reference 6, "FSAR Section 15.4". Delete it from the Reference List if not applicable.

Response to Question 16-91:

The reference in Technical Specification Section 3.7.1 Bases to U.S. EPR FSAR Tier 2, Section 15.4 will be deleted.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.1 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-92:

TS Bases B 3.7.3, Main Feedwater Valves (MFVs).

EPR design has 3 Main Feedwater (MFW) valves in-series which can perform the isolation function for each respective full load feedwater flow paths. With 2 valves inoperable in Condition B, single failure criteria cannot be afforded by the remaining valve. With 3 valves inoperable in Condition C, the loss of the isolation function is eminent. The justifications in the bases for Required Actions A1, B1 and C1 are exactly the same although Condition A is for one valve inoperable, Condition B is for two valves, and Condition C is for three valves inoperable.

Clarify the availability of redundant valves when Condition B or C is entered. Revise the TS bases B 3.7.3 as appropriate.

This is required to ensure the supporting information in TS bases accurately refects requirements specified in the TS.

Response to Question 16-92:

As discussed in U.S. EPR FSAR Tier 2, Chapter 10, Section 10.4.7, the U.S. EPR design has three valves in series which perform the isolation function. The Conditions and Required Actions in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.3 recognize this added redundancy. These LCO Conditions were compared to the STS (NUREG-1431), Section 3.7.3.

Condition A addresses the situation where one of the three full load main feedwater (MFW) valves is inoperable. In this condition, the STS LCO is fully met. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.3 Condition A imposes a 7-day AOT.

Condition B addresses the situation where two of the three MFW full load valves are inoperable. This condition is analogous to STS Condition B in that one isolation valve remains operable. A 72 hour Completion Time is provided to restore two valves in the flow path to OPERABLE status or isolate the flow path.

Condition C addresses all three MFW full load flow path valves being inoperable. This condition is analogous to STS Condition D in that all isolation valves in the flow path are inoperable. Eight hours is allowed to isolate the flow path or return one valve in the flow path to OPERABLE status.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.3 Conditions B and C and Bases will be revised to clarify the MFW flow path requirements.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.3 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-93:

TS Base B 3.5.2, ECCS - Operating.

Provide additional discussion and justification regarding the available capacity of ECCS flow when "all 4 ECCS trains are Operable" to when "less than 100% ECCS flow is available." The Bases, Condition B, states that these Conditions match the assumptions in FSAR, Chapter 15 (when an additional ECCS train is out of service). This is not apparent when the ECCS trains are concurrently in Conditions B and C, especially if a single failure were to occur, when this configuration appears to be outside of the analyses. This additional information needs to be added to the Bases to ensure an accurate and clear understanding of the information.

In addition, the Bases, LCO discussion, states that "four 100% capacity independent (cross-connect closed) ECCS trains are required to ensure that sufficient ECCS flow is available." Expand upon this statement at each degraded Condition, for cross-connects open and then closed, so that it can be demonstrated that the remaining Operable ECCS trains can respond successively to the postulated accident in addition to the assumed unavailability of other ECCS trains. The information is required to ensure the completeness and accuracy of the Conditions specified for the proposed LCO statement.

Response to Question 16-93:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 covers a range of ECCS operability from four emergency core cooling system (ECCS) [medium head safety injection (MHSI) and low head safety injection (LHSI)] trains operable (LCO 3.5.2) through progressive levels of degradation down to the point of having less than two equivalent ECCS trains available (Condition E).

U.S. EPR FSAR Tier 2, Chapter 15 assumptions for a bounding large break loss of coolant accident (LBLOCA) are that one ECCS train is unavailable due to maintenance, one ECCS train is lost due to the single failure of one diesel, and one ECCS train is lost to the break. One ECCS train is adequate to accomplish the safety function of cooling the core.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Conditions A and B address the inoperability of one MHSI train or one LHSI train, respectively. A short time period is allowed to open the LHSI cross connects to add protection for specific break locations as discussed in U.S. EPR FSAR Tier 2, Section 6.3.2.1. The 120 day allowed outage time (AOT) is discussed in response to Question 16-102.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Condition C will be revised to address two ECCS trains inoperable rather than two MHSI or two LHSI trains inoperable. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Bases will be revised to add clarification for the ECCS train requirements.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 Condition C and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-94:

TS Bases B 3.5.4, In-Containment Refueling Water Storage Tank.

Provide additional information to include the IRWST water temperature operability requirements that the must be met in the Bases discussion for this LCO.

The LCO does require water temperature limits to met for operability in Modes 1, 2, 3, and 4, however, the Bases discussion for this LCO does not include IRWST water temperature as a requirement for Operability. In addition, the Bases for Action A.1 does not include this same IRWST water temperature in the three places required.

The additional information will be used to ensure the Bases are complete and accurate for this LCO.

Response to Question 16-94:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.4 Bases will be revised to include in-containment refueling water storage tank (IRWST) temperature as a requirement.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.5.4 will be revised as described in the response and indicated on the enclosed markup.

Question 16-95:

TS Bases B 3.5.4, In-Containment Refueling Water Storage Tank.

Provide additional information to clarify or revise the Bases, Applicable Safety Analyses, Actions for B.1, and the Bases, SR 3.5.4.2 to describe the effects of the the IRWST exceeding a borated water volume of 523,703 gallons. Justify Condition B not explicitly stating that the water volume is not within limits. Confirm that extra IRWST water volume does not inhibit response time and is this only parameter that determines switching to hot leg recirculation following a LOCA.

In addition, provide additional information that explains the methodology for using the FSAR Table 6.3-4 required design total volume of minimum and maximum water in the IRWST in cubic feet to meet the SR 3.5.4.2 which is expressed in gallons. Identify the installed instrumentation, including its accuracy and MCR display(s) that will be used to ensure this FSAR requirement is met.

This additional information will be used to ensure completeness and accurate of this LCO. It will also ensure that the Conditions specified are consistent with the Actions to restore Operability.

Response to Question 16-95:

The in-containment refueling water storage tank (IRWST) will be maintained with a contained water volume between 500,342 gallons and 523,703 gallons in accordance with U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.5.4.2. An actual level above 523,703 gallons is not a safety concern. As noted in the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.4 Bases, the maximum volume was an input into the post-loss of coolant accident (LOCA) boron precipitation calculation. This does not imply that the IRWST volume impacts the response time for switchover to hot leg recirculation. Condition A will be revised to include the IRWST volume. This change is consistent with AP-1000 plant Technical Specifications.

U.S. EPR FSAR Tier 2, Chapter 6, Table 6.3-4 will be revised to provide an equivalent gallon requirement.

The following displays are available to verify IRWST inventory:

a) Instrumentation used—30JNK10/11 CL050 / CL052.

b) Display—General Arrangement Display, IRWST Display, and SICS.

Instrumentation accuracy is specific to procured equipment and these cannot be specified at this time.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 6, Table 6.3-4 will be revised as described in the response and indicated on the enclosed markup.

Question 16-96:

TS 3.5.5, Extra Boration System (EBS).

Provide an explanation for the exclusion of a SR to verify that the applicable valves automatically reset after the EBS containment isolation valves close, following a containment isolation signal, to permit manual opening in the MCRB.

This additional information will be used to ensure complete and accurate Surveillance Requirements are provided.

Response to Question 16-96:

An surveillance requirement (SR) will be added to the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 to verify that the applicable valves automatically reset after the extra borating system (EBS) containment isolation valves close, following a containment isolation signal, to permit manual opening from the main control room.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-97:

TS 3.5.5, Extra Boration System (EBS).

Provide additional information to justify the parameters for "temperature and volume limits not being within limits" not having an unique Condition that is separate from the EBS pump or MOV valve being inoperable. In addition, discuss the anticipated length of times to restore Operable status for the various problems that may occur within EBS. This discussion should be suitable for inclusion in the Bases for Actions to clearly justify the Completion Time (CT) specified.

This additional information will be used to ensure all of the Conditions specified are consistent with the Actions needed to restore LCO Operability or to exit the proposed LCO Applicability.

Response to Question 16-97:

There is a unique condition that requires separation of the two mechanical trains of extra borating system (EBS) [EBS pump or motor operated valve (MOV)] from the borated water. This requires 100 percent of “The volume of concentrated boric acid required to maintain subcritical is divided between the two EBS tanks” to be maintained for the plant’s shutdown requirements. With temperature and volume not within limits, sufficient boron may not be available to maintain the core subcritical, if a reactor coolant system (RCS) cooldown is required.

One possible cause of the EBS tank temperature decreasing below the limit is low outside ambient temperature coupled with a failure in the safety-related portion of the fuel building ventilation system (FBVS) to maintain required EBS room temperatures. The margin for EBS tank temperature to decrease before boron precipitation begins allows a reasonable time to either restore the FBVS or start the EBS pump to allow recirculation to heat the EBS tank within required limits. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 Condition A LCO completion time provides for 72 hours to restore the parameter within limit.

Other possible causes of the EBS tank volume decreasing below limits are: leakage, sampling, or testing. The margin for EBS tank volume would make it unlikely that such a decrease in volume would occur without detection. Make-up to the boric acid can be supplied from the reactor boron water makeup system (RBWMS). The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 Condition A LCO completion time provides for 72 hours to restore the parameter within limit.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for Section 3.5.5 Actions A.1, B.1 and C.1 will be revised to provide additional detail.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.5.5 will be revised as described in the response and indicated on the enclosed markup.

Question 16-98:

TS 3.5.5, Extra Boration System (EBS).

Provide justification for selecting the 12-hour Completion Time (CT) for Required Action D.1 to place the plant in Mode 3 from a higher Mode 1 or Mode 2. Revise TS 3.5.5 and related information in the bases, as appropriate.

The 12-hour CT is not consistent with the historical 6 hours for bringing the plant from Mode 1 to Mode 3 and in other similar situations within the EPR TS, e.g., TS 3.4.4 .

Response to Question 16-98:

The Completion Time for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5, Action D.1 and Bases will be changed to 6 hours.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-99:

TS Bases B 3.5.5, Extra Boration System (EBS).

Provide a description of maintaining or changing the temperature of the EBS water if there is no pipe heat tracing and an explanation of influence from the fuel building ventilation system on the EBS water temperature. Confirm that the ventilation system is safety-related. Provide the methodology for directly monitoring and verifying the water temperature in SR 3.5.5.1. Revise the TS and bases as appropriate to include this information.

The additional information will be used to ensure that accuracy and completeness of this SR

Response to Question 16-99:

The extra borating system (EBS) is designed to prevent boric acid crystallization during standby without heat tracing. The temperature inside the Fuel Building where the tanks and main lines are located is maintained at or above 68°F by safety-related portions of the fuel building ventilation system (FBVS). For piping outside the temperature-controlled rooms, piping is filled with lower boron concentration water from the in-containment refueling water storage tank (IRWST) to prevent crystallization.

The margin for EBS tank temperature to decrease before boron precipitation begins allows a reasonable time to either restore the FBVS or start the EBS pump to allow recirculation to heat the EBS tank. The temperature of the EBS water is monitored by the safety-related ventilation systems and the EBS tank is monitored directly with a temperature probe.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5.1 and Bases will be revised to add the EBS pump room requirement and discussion.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-100:

TS 3.5, Emergency Core Cooling Systems.

Correct the following editorial errors:

1. Revise LCO 3.5.3, Required Action B.1, to correctly state Mode 5 as "MODE 5".
2. Revise LCO 3.5.5 or FSAR 6.8 to adopt a consistent name for the Extra "Boration" System or the Extra "Borating" System.

Response to Question 16-100:

1. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.5.3 will be revised to correctly state Mode 5 as “MODE 5”.
2. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.5.5, Table 3.3.2-1, and Bases Sections 3.1.1, 3.3.2, and 3.5.5 will be revised to consistently reference “Extra Borating System”, where applicable.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.5.5, Table 3.3.2-1, and Bases Sections 3.1.1, 3.3.2, and 3.5.5 will be revised as described in the response and indicated on the enclosed markup.

Question 16-101:

TS 3.5.5, Extra Boration System (EBS).

Provide the additional information to justify the proposed Frequency of a "Staggered Test Basis" in the Bases for SR 3.5.5.7 or provide a different frequency and justification.

Typically, the "Staggered Test Basis" is to allow testing one subsystem, train, or designated components at a time during the interval specified by the Surveillance Frequency when the system has a large number of subsystems or components or if one of the systems need to be kept in service . The Extra Boration System (EBS) in the EPR design is a simple two-train system with only a few components that require manual actuation to perform their design function.

This additional information will be used to ensure complete and accurate Surveillance Requirements are provided.

Response to Question 16-101:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.5.5.7 will be revised to delete the allowance of "Staggered Test Basis" testing.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.5 will be revised as described in the response and indicated on the enclosed markup.

Question 16-102:

TS Bases B 3.5.2, ECCS - Operating.

Provide a technical justification for the proposed 120 day Completion Time (CT) for restoring one MHSI pump to operable status for incorporation into the Bases for Action A.1. In addition, the Bases for Action B.1 and B.2 does not match the Condition B Actions when a 60 day CT is proposed and a 120 days CT is stated.

There are no industry standards or regulatory guidance that permits safety-related equipment to remain inoperable for this amount of time. Industry guidance for Risk Managed Technical Specifications (RMTS), Initiative 4B, as accepted by the staff in the Safety Evaluation of NEI 06-09, NUCLEAR ENERGY INSTITUTE (NEI) TOPICAL REPORT (TR) NEI 06-09, "RISK-INFORMED TECHNICAL SPECIFICATIONS INITIATIVE 4B, RISK-MANAGED TECHNICAL SPECIFICATIONS (RMTS) GUIDELINES", (ADAMS Accession Number ML071200238) requires a backstop of 30 days for risk based extensions to TS Completion Times.

The information will be used to ensure that all operating requirements and restrictions of the Actions are compatible with Completion Times that do not invalidate assumptions of the analyses in the FSAR.

Response to Question 16-102:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2, Condition B, allowed outage time (AOT) for one train of the low head safety injection (LHSI) subsystem inoperable will be revised from 60 days to 120 days.

The details regarding the safety analysis and assumptions regarding U.S. EPR operation with three operable emergency core cooling system (ECCS) trains and three emergency diesel generators (EDG) with the alternate feed aligned are provided in U.S. EPR FSAR Tier 2, Chapter 15 and Chapter 8, respectively. Excerpts are provided below:

- With regards to the safety analysis, the assumptions regarding EDG availability are included, as appropriate, with each event. For example, U.S. EPR FSAR Tier 2, Section 15.6 addresses decrease in reactor coolant inventory events. U.S. EPR FSAR Tier 2, Section 15.6.1.2, *Methods of Analysis and Assumptions*, states:

  "The most reactive control rod is assumed not to insert at reactor trip (RT). A Loss of Offsite Power (LOOP) is assumed to occur with RT. Subsequent to an RT, the limiting single failure is taken as the failure of one emergency diesel generator (EDG), resulting in the unavailability of one train of pumped SIS (MHSI, LHSI, and EFWS). A second EDG is assumed to be under maintenance and therefore unavailable, causing a second train of pumped SIS to be unavailable."

- The electrical aspects are described in U.S. EPR FSAR Tier 2, Chapter 8. For example, U.S. EPR FSAR Tier 2, Section 8.1.2, *Onsite Power System Description*, states:

  "An alternate feed is provided from EPSS division 2 to EPSS division 1 to supply a standby source of power to required safety related systems, safety-related support systems, or components that do not have four 100 percent redundant trains when the EPSS division 1 EDG or certain portions of the EPSS division 1 electrical distribution system are not available. A similar alternate feed provides standby power to EPSS division 2, from EPSS division 1 when the EPSS division 2 EDG or certain portions of the EPSS division 2 electrical distribution system are not available. Similar alternate feeds are used between EPSS division 3 and EPSS division 4. Implementation of the alternate feed is completed manually to satisfy single failure criteria when certain electrical components, including EDGs, are out of service.

Each EDG automatically starts and connects to its EPSS 6.9 kV switchgear when a loss of power or a degraded voltage condition is detected at the respective division supply bus. An automatic start will also occur if a safety injection signal (SIS) is initiated from the protection system. The required safety-related loads are automatically sequenced onto the EDG when the generator has obtained nominal speed and voltage, and a loss of voltage or a degraded voltage signal is received. Each EDG has the capacity and capability to power the required safety-related loads when an alternate feed is implemented between divisions."

- U.S. EPR FSAR Tier 2, Section 8.3.1.1.1, *Emergency Power Supply System*, states:

  "EPSS divisions are functionally independent and physically separated from the others during normal bus alignments. An alternate feed is provided between EPSS divisions 1 and 2 (first divisional pair) to provide the normal and standby source of power to required safety-related systems, safety-related support systems, or components that do not have the required redundancy when certain electrical components, including the division 1 emergency diesel generator (EDG), are out of service. A similar alternate feed provides standby power to EPSS division 2, from division 1 when certain electrical components, including the division 2 EDG are out of service. Similar alternate feeds are used between divisions 3 and 4 (second divisional pair)."

- U.S. EPR FSAR Tier 2, Section 8.3.1.2.4, *Compliance with GDC 17*, states:

  "The EPSS has four divisions, normally powered from the preferred power source, each with an independent and redundant EDG assigned to their respective switchgear 31BDA, 32BDA, 33BDA, and 34BDA. The EPSS divisions combine to make two divisional pairs. Division 1 and 2 constitute the first divisional pair while divisions 3 and 4 constitute the second divisional pair. The EPSS safety-related loads are separated between the divisional pairs and a loss of one divisional pair will not prevent the minimum safety-related functions from being performed.

  The four EPSS divisions are normally functionally independent and physically separated from each other. During periods a standby power source is out of service, or other similar maintenance activities, alternate feeds are provided between Division 1 and Division 2 or between Divisions 3 and Division 4 as appropriate for the out-of-service EDG. The alternate feed configuration, consistent with separating the safety related loads between divisional pairs, maintains the plant capability to complete safety-related functions coincident with a single failure."

With regard to risk insights, the 120 day AOT is not a risk-informed value. Since the U.S. EPR safety analysis assumptions are satisfied with three operable ECCS trains and three EDGs with the alternate feed aligned, an indefinite AOT is justifiable from a deterministic standpoint. The proposed AOT does not constitute a temporary relaxation of the requirement for a postulated single failure concurrent with a postulated accident or anticipated operational occurrence (AOO). Therefore, the proposed 120 day AOT does not constitute a deviation from the Standard Technical Specifications. Rather, it is an additional conservative restriction that maximizes the availability of the EDGs. In general, a licensee would not be expected to submit risk information in support of a position that is consistent with currently approved staff positions. These licensee positions are normally evaluated by the staff using traditional engineering analyses.

AREVA's position is consistent with that proposed by other DC applications:

- ESBWR Technical Specifications (May 2008) for Distribution Systems - Operating (LCO 3.8.6) requires only three of the four divisions of DC and Uninterruptible AC Electrical Power Distribution buses 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 Design Basis Accident (DBA).

- APWR Technical Specifications (December 2007) for Safety Injection System (SIS) - Operating (LCO 3.5.2) requires only three of four SIS trains to be operable.

With regards to the staff position that there are no industry standards or regulatory guidance that permits safety-related equipment to remain inoperable for this amount of time, Technical Specification AOT for plants which lack the redundancy of the U.S. EPR or other next generation plants have previously been acknowledged as temporary relaxations of the single failure criteria for specified and limited periods of time. It is not appropriate to apply that logic to this case. With regards to guidance on how the U.S. EPR should treat availability of equipment that is above and beyond that required by the safety analysis. AREVA bases its position upon:

a) 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, and

b) Regulatory Guide 1.160, Revision 2, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, which endorses Revision 2 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants".

This guidance would be required to perform maintenance activities including, but not limited to: TS Surveillance Requirements, post-maintenance testing, and corrective or preventative maintenance, for equipment covered by the proposed 120 day AOT. In part, the NUMARC guidance states:

– Performance of maintenance during power operations should be planned and scheduled to properly control out-of-service time of systems or equipment.

– The assessment method may use quantitative approaches, qualitative approaches, or blended methods. In general, the assessment should consider:

- The degree of redundancy available for performance of the safety function(s) served by the out-of-service SSC.

Neither the guidance or guideline specify maximum timeframes nor preclude the proposed AOT for the equipment.

The level of detail contained in the Bases for this completion time is consistent with the level of detail provided for completion times throughout the Bases, both within the U.S. EPR Generic Technical Specifications and the Standard Technical Specifications for Westinghouse Plants.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.2 will be revised as described in the response and indicated on the enclosed markup.

Question 16-103:

TS Bases B 3.9.5, Residual Heat Removal (RHR) Loops - Low Water Level.

Provide additional information to explain a "Note" added to the EPR GTS.

Provide additional information to explain/justify the note added to the GTS Bases SR 3.9.5.2 that states that the SR is "not required to be performed until 24 hours after a required RHR loop is not in operation."

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-103:

A response to this Question will be provided by February 19, 2009.

Question 16-104:

TS Bases 3.9.3, Containment Penetrations.

Provide additional information to explain differences between the EPR Bases and the applicable STS Bases regarding TSTF-51, Rev 2, and TSTF-471, Rev 1.

The EPR Bases, Section B 3.9.3, Applicable Safety Analyses, 4th sentence does not contain consistent wording with the STS, Section B 3.9.3, Applicable Safety Analyses, explaining the changes of TSTF-51, Rev 2 and TSTF-471, Rev 1. It appears the sentence is intended to read as, "in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability or a minimum decay time of [x] days without containment closure capability, ensures that the release." A technical justification for the differences between the EPR GTS and the STS should include a discussion on how the departure from a fraction of 10 CFR 100 limit to the proposed guidance in Regulatory Guide 1.1.83 has been conducted within EPR GTS to ensure consistency. Provide a comparison where the previous requirements have been uniformly replaced by the new requirements. Provide a a technical justification for the intent of the different wording or applicable changes to make the EPR Bases consistent with the STS Bases regarding TSTF-51, Rev 2, and TSTF-471, Rev 1.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases

Response to Question 16-104:

The description of a fuel handling accident is provided in U.S. EPR FSAR Tier 2, Chapter 15, Section 15.0.3.10. A fuel handling accident was evaluated to occur in an open containment or fuel building with a single fuel assembly that has decayed a minimum of 34 hours. The analysis assumptions are provided in U.S. EPR FSAR Tier 2, Chapter 15, Table 15.0-46. As noted in this table, no credit is taken for a closed containment, or fuel building or for a reduction by the exhaust filtration systems.

The current U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.9.3 and Bases address handling of recently irradiated fuel. The U.S. EPR conservatively assumes a decay time of 34 hours before fuel movement. Since unit cooldown, reactor vessel head removal, and other necessary activates that proceed fuel movement are estimated to take approximately 90 hours, U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.9.3 Limited Condition for Operation (LCO) and Bases are not needed and will be deleted. As provided in the response to RAI Question 16-65, a new Technical Specification LCO and Bases will be added to limit irradiated fuel movement to > 34 hours.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.9.3 LCO and Bases will be deleted as described in the response and indicated on the enclosed markup.

Question 16-105:

TS Bases 3.9.3, Containment Penetrations.

Provide additional information to explain the differences between the EPR Bases and the applicable STS Bases.

The EPR Bases, SR 3.9.3.2, second paragraph does not contain the text from the Westinghouse STS Bases, SR 3.9.3.2, that describes the justification for selection of the 7 days Frequency. The STS Bases text in question states that "[a] surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO." In addition, the EPR Bases, SR 3.9.3.2, second paragraph, last sentence is missing the reference used to identify the radioactivity dose limit to the environment at the EAB.

This additional information is needed to ensure the accuracy and completeness of the EPR Bases.

Response to Question 16-105:

See the response to question 16-104.

FSAR Impact:

See the response to question 16-104.

Question 16-106:

TS Bases 3.9.3, Containment Penetrations.

Provide additional information to explain differences between the EPR Bases and the applicable STS Bases.

The EPR Bases, Section B 3.9.3, Background Section, 6th & 8th paragraphs refer to the plant specific name of the flow purge system / penetration as "partial" flow herein. The EPR GTS, LCO 3.6.3, refers to the same as "low" flow.

The EPR Bases, Section B 3.9.3, Background Section, the last paragraph states that an equivalent isolation method requires approval but does not identify the approving authority. In addition, according to the Westinghouse STS Bases, it appears that this option has only been approved for GPU Nuclear under SE-0002000-001, Rev. 0, May 20, 1988. Justify omitting or not updating this reference in EPR GTS.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases

Response to Question 16-106:

See the response to question 16-104.

FSAR Impact:

See the response to question 16-104.

Question 16-107:

TS Bases 3.9.2, Nuclear Instrumentation.

Provide a technical justification for the exception that is added to the EPR GTS, LCO 3.9.2, Action A.1.

The EPR GTS, LCO 3.9.2, Action A.1 has an exception not included in Westinghouse STS that requires licensees to "suspend positive reactivity additions, except the introduction of coolant into the RCS." Provide a technical justification for this exception There is no stated operational basis for needing to continue inventory additions when the Bases for Action A.1 clearly state restrictions for RCS inventory additions and avoidance as the preferred Action. With the combined technical justification of Action A.1 & A.2 in this same paragraph, the related EPR Bases are confusing (especially the 4th and 5th sentences), and should be rewritten separately to address each Action appropriately. The EPR Bases description for B.1 and B.2 should be separated, as well.

This additional information is to ensure the accuracy, completeness and clarity of the EPR GTS and Bases.

Response to Question 16-107:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.9.2, Action A.1 will be revised to delete the noted exception. Actions B.1 and B.2 will be separated. These changes will make the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications and Bases consistent with approved TSTF-471, Revision 1.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.9.2 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-108:

TS 3.7.2, Main Steam Isolation Valves (MSIVs).

Justify the removal of the TS limiting closure times from SR 3.7.2.3. Confirm that the correct closure time verified is the same as assumed in the FSAR. Also SR 3.7.2.2 and SR 3.7.2.3 have a SR Note with an inadequate justification in the Bases that consists of repeating the same words of the Note. Correct the justification for the SR Note.

This information will be used to complete the review and content of the SR and the associated Bases.

Response to Question 16-108:

The main steam isolation valve (MSIV) closure time was deleted from the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.7.2.3 in accordance with approved TSTF-491, Revision 0.

The partial stroke test is intended to demonstrate free movement of the MSIVs under normal steam pressure/flow conditions. Performing this surveillance in Mode 3 would not be meaningful since operational steam conditions are not present in Mode 3.

U.S. EPR FSAR Tier 2, Section 10.3.2.3 describes the startup of the main steam system. Heatup of the Reactor Coolant System and Main Steam System can be accomplished with the MSIVs open or closed. If the heatup is performed with the MSIVs closed, the downstream piping is slowly warmed once there is adequate steam pressure using the main steam warming control valves. Performance of a partial stroke test of the MSIVs during this evolution would not be representative of normal operating conditions, nor would be desirable because this could produce an operational transient or an upset plant condition.

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.7.2.3 Note is consistent with NUREG-1431, Revision 3.1.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-109:

TS 3.7.15, Spent Fuel Storage Pool Boron Concentration.

Provide a technical explanation for LCO 3.7.15 and justify including "the boron enrichment shall be > or = 37%" as a parameter that must be part of the LCO statement.

FSAR 9.1.2.2.2 lacks sufficient detail as does the supporting Base proposed for this LCO. There is inconsistency in reference to the B10, B-10, or none.

This information will be used to complete the review and content of the LCO.

Response to Question 16-109:

The U.S. EPR uses enriched boric acid (≥ 37%) for reactivity control. Since this is a higher occurrence of B-10 than occurs in natural boric acid (approximately 20% B-10) it is necessary to maintain the concentration and isotopic concentration of B-10 in the spent fuel pool. This approach is consistent with surveillance requirements for the accumulators U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.5.1.6, SR 3.5.4.4, and SR 3.5.5.6.

The title of U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.15 and Bases will be changed to "Spent Fuel Pool Boron Concentration and Enrichment."

The B-10 notation in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.5.1, 3.5.4, 3.5.5, and Section 3.7.15 and Bases will be changed to be consistent with the U.S. EPR FSAR.

Use of enriched boric acid in the spent fuel pool is discussed in U.S. EPR FSAR Tier 2, Section 9.1.2.2.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.1, 3.5.4, 3.5.5, and Section 3.7.15 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-110:

TS 3.7.12, Safeguard Building Controlled Area Ventilation System.

Explain the difference in the requirements for the SR 3.7.12.7 test flow rate as < and = 2640 cfm versus the Bases SR discussion as < and = 2400 cfm versus the test range in VFTP of TS Section 5.5.10.

The information will be used to ensure Operability of the required SBVS components.

Response to Question 16-110:

The text in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.7.12.6 and 3.7.12.7 Bases will be revised to reflect the correct flow of ≤ 2640 cfm which is consistent with the test range given in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.5.10.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.12 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-111:

TS 3.7.10, Control Room Emergency Filtration (CREF).

Explain the purpose of the Reviewer's Note on the optional adoption of Required Action for Condition D for toxic gases response in TS 3.7.10.

The information will be used to understand operation of the CREF and what must be included in the Bases to support the proposed LCO and Action statements.

Response to Question 16-111:

Refer to the response to RAI 89 question 6.2.3-1.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-112:

TS 3.6.7, Annulus Ventilation System (AVS).

Provide a new Surveillance Requirement to ensure the normal operation filter train safety-related annulus air inlet and exhaust motor operated isolation dampers are tested to verify Operability. Provide the appropriate Conditions for inoperability and Bases to fully describe the safety feature being tested.

For those components that must continue to function or reposition for accident operation, appropriate TS requirements must be provided.

Response to Question 16-112:

An Action and Surveillance requirement will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.7 and Bases for the isolation dampers.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.7 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-113 :

TS Bases B 3.6.7, Annulus Ventilation System.

Provide additional explanation of the design and operation of the AVS trains to resolve apparent differences between the LCO, Bases and FSAR for the following items:

Confirm that there is one normal operation filtration train and that one system ensures shield building integrity during normal operations. Confirm that there are no shared components between the normal operation filtration train and the two accident filtration trains. Confirm that the safety-related annulus air inlet and exhaust motor operated isolation dampers of the normal filtration train must function or close to seal the secondary containment boundary and to assure the accident filtration trains can perform their intended function.

When the non-safety related normal operation filtration train is not available or is inoperable, provide the operational effects on the safety-related portions of the AVS and LCO Conditions that would be entered. Confirm that there is a control room manual start for both the normal operation and the accident filtration trains of the AVS. In the Bases Background text for the fifth sentence of the sixth paragraph, compare the main HEPA filter bank to the downstream & upstream filter banks.

The Bases need to be correct, the planned operation of the trains must be stated and in agreement with the LCO requirements.

Response to Question 16-113:

As discussed in U.S. EPR FSAR Tier 2, Section 6.2.3.2.2, one non-safety train is normally in operation and maintains a negative pressure in the annulus, maintaining Shield Building integrity during normal operation.

The normal operation filtration train is shown on U.S. EPR FSAR Tier 2, Chapter 6, Figure 6.2.3-1 and the two accident filtration trains on U.S. EPR FSAR Tier 2, Chapter 6, Figure 6.2.3-2.

The four safety class motor-operated air tight dampers isolate the annulus from the non-safety normal operation train in case of a design basis accident (DBA). The two isolation dampers in both the supply and exhaust trains are powered by separate divisions and are supplied by the emergency diesel generators (EDG). Each isolation damper can be operated either automatically or manually from the Main Control Room (MCR).

In normal operation mode, if there is a loss of negative pressure in the annulus, failure of the nuclear auxiliary building ventilation system (NABVS), or loss of offsite power (LOOP), the normal operation filtration train is considered lost and one of the accident filtration trains is switched on. The two isolation dampers on both the normal supply and exhaust trains are closed and one of the two accident filtration trains is switched on. The other accident filtration train is available for backup. Since the non-safety train is not credited, no LCO would be entered.

The isolation dampers on the normal operation filtration train and the accident filtration trains can be operated either automatically or manually from the MCR. Refer to U.S. EPR FSAR Tier 1, Chapter 2, Table 2.6.3-1.

The upstream high efficiency particulate air (HEPA) filter removes the fine discrete particulate matter from the air stream. The downstream HEPA filter is in-line following the charcoal adsorber.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.7 and Bases will be revised to add clarification.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.7 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-114:

TS 3.6.6, Shielding Building.

Provide a justification for the following difference. In SR 3.6.6.4, it is required to verify the negative pressure limit can be maintained at a flow rate of < 1295 cfm. In the Bases for SR 3.6.6.3 and SR 3.6.6.4, the negative pressure limit can be maintained at a flow rate for one AVS train of < 1320 cfm.

Provide a clarification for the basis for the "selection of the annulus pressure" in the second sentence of the paragraph of the Bases for SR 3.6.6.3 and SR 3.6.6.4.

The Bases need to be correct and in agreement with the LCO requirements.

Response to Question 16-114:

The Bases for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.6.3 and 3.6.6.4 will be revised to reflect a flow rate of 1295 cfm.

The negative pressure assumed in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.6.3 and 3.6.6.4 (0.25 inches water gauge) is the NRC required negative pressure from SRP 6.2.3 Section II SRP Acceptance Criteria, 3.B.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.6 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-115:

TS Bases 3.6.5, Containment Air Temperature.

Provide an explanation for inclusion into Bases SR 3.6.5.1 of methodology for developing and calculting "weighted average" and justify its preference over other methods such as arithmetic average of instrument readings.

This information is intended to ensure that the Bases statement has not introduced a difference from the initial starting points for analysis assumptions as presented in the FSAR.

The information will be used to determine whether the initial temperature point assumed in the analysis change is affected by the re-calculated reading or not.

Response to Question 16-115:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.5.1 Bases will be changed to reflect an arithmetic average.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.5 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-116:

TS Bases 3.6.3, Containment Isolation Valves.

Revise the Applicable Safety Analyses of the Bases to justify indicating in the last sentence of the fifth paragraph that the low flow purge valve design precludes a single active failure from compromizing the containment boundary (e.g. having a diverse power source for the inboard and outboard isolation valves to preclude common mode failure). Furthermore, the sixth paragraph is incomplete as "10 CFR 50.36(c)(2)(ii)" is missing.

Confirm that the Low Flow and Full Flow purge valves have resilient seals. Identify the Condition entered when a low flow purge valve is not within its leakage limit.

Revise the LCO discussion in the second sentence of the second paragraph to justify the status of the Full Flow Purge valves as sealed closed. Revise the ACTION discussion in the first paragraph to remove the last sentence regarding opening allowance stated in SR 3.6.3.1 which dos not exist. Also, remove Reference #5 which is no longer mentioned in the bases discussion of SR 3.6.3.1.

Response to Question 16-116:

As discussed in U.S. EPR FSAR Tier 2, Sections 6.2.4.2.5 and 9.4.7.3, the containment purge inboard and outboard isolation valves are provided with separate power supplies and controls. The Applicable Safety Analysis discussion in the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.6.3 Bases will be revised to clarify these features. Also, the reference to 10 CFR 50.36(c)(2)(ii) will be added.

The Low Flow and Full Flow purge valves are not the typical butterfly valves used in containment purge systems in many operating nuclear plants. U.S. EPR containment purge isolation valves are pneumatically actuated and spring closed, leak-tight valves with resilient seals.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.3.6 will be revised to add the Low Flow Purge Valves.

The U.S. EPR containment purge isolation valves are designed to close in the event of a loss of coolant accident (LOCA) or main steam line break (MSLB) inside containment. Therefore the NUREG-1431 restriction to "seal" the full flow purge valves closed due to their inability to close against LOCA conditions is not applicable. However the full flow purge valves are maintained sealed closed in Modes 1, 2, 3, and 4 as assumed in the design basis accident (DBA) dose analysis. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.3.1 will be revised to add "sealed" closed. Also, U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.3.1 will be revised to add the exception for Condition D. Additional text will be added to the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.3.1 Bases that incorporates Reference 5.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.3 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-117:

TS Bases 3.6.1, Containment.

Clarify, expand, and reconcile the containment leakage limits stated in the Applicable Safety Analyses Bases 3.6.1 with the FSAR Tier 2, 6.2.6.1 requirements.

For the Type A test, clarify that the allowable leakage rate of 0.25% is an administrative leakage rate, or is it the allowable leakage rate versus an apparent in-service design leakage rate of 0.75 La at peak pressure. Furthermore, clarify if the stated units of "(percentage) containment air weight day" or "(percentage) containment mass per day" introduces any differences here or need for conversion factors. Explain the interpretation of the rest of the LCO paragraph using the explanation provide above for which leakage limits apply at first start-up.

The testing criteria assumed in the analyses must be consistent with the in-service permissible limits verified of the containment boundary under TS 5.5.15.

Response to Question 16-117:

As noted in U.S. EPR FSAR Tier 2, Chapter 15, Tables 15.0-40 and 15.0-50, the assumed containment leakage rate is 0.25 weight percent per day for the 0–24 hour period. This is the leakage value stated in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.5.15c as L_a at P_a. The containment leakage limits for preservice and inservice Type A tests are provided in U.S. EPR FSAR Tier 2, Section 6.2.6.1. A revision will be made to U.S. EPR FSAR Tier 2, Section 6.2.6.1 to change "containment air mass" to “containment air weight".

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.6.1 and Section 5.5.15 are consistent with the STS.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 6.2.6.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-118:

TS Bases 3.7.2, Main Steam Isolation Valves (MSIVs).

Provide additional information to explain the new staggered SR frequency for the EPR GTS, Plant Systems.

The EPR GTS, SR 3.7.2.4 Frequency is 24 months on a "STAGGERED TEST BASIS for each MSIV pilot valve." The EPR Bases should provide a more detailed description of this new staggered frequency to include a more precise testing method and representative schedule required for the proposed SR. Clarify the expected testing performed for each pilot valve in each control line at each interval. Confirm that this test inclusively initiates a full cycle of each MSIV.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-118:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for SR 3.7.2.4 will be revised to clarify that each main steam isolation valve (MSIV) is cycled through a full closure on a 24 month Frequency. The test will alternate use of the redundant control lines.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.2 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-119:

TS 3.7.3, Main Feedwater Valves (MFVs).

Provide a technical justification for excluding the limiting closure time for SR 3.7.3.1 or revise the SR to include closure times that are consistent with the FSAR.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-119:

The Main Feedwater Valves closure times were excluded from U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.7.3.1 in accordance with approved TSTF-491, Revision 0.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-120:

TS Bases 3.7.8, Essential Service Water System (ESWS).

Provide additional information needed to explain the two "Notes" in the EPR GTS, LCO 3.7.8, Action Section.

The EPR Bases, Section b 3.7.8, Action Section, with regards to the two "Notes" in the EPR GTS, LCO 3.7.8, Action Section, requires additional information to adequately provide the bases/reasons for the notes because it currently just repeats the contents of the Notes ; rather than a justification for the affect of the entry into the Action for the EPR ESW System design. Also, explain design differences and locating the Notes here; rather than previously, when located as notes only Applicable to a Required Action A.1.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-120:

The Actions Section for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Section 3.7.8 Bases contains the same information and level of detail provided in the Actions Section in the Standard Technical Specifications for Westinghouse Plants NUREG-1431, Bases 3.7.8 "Service Water System (SWS)".

The Notes have been relocated as per TSTF-GG-05-01, “Writer’s Guide For Plant-Specific Improved Technical Specifications”, §4.1.6 where the Notes are applicable to all Conditions and Required Actions. The Conditions and Required Actions for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Bases 3.7.8 "Essential; Service Water (ESW) System" have Completions Times that are greater than the Completion Times of the annotated supported systems.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-121:

TS Bases B 3.7.9, Safety Chilled Water (SCW) System.

Provide additional information needed to explain and justify the Action "Note" added to LCO 3.7.9.

The EPR GTS, LCO 3.7.9, has a new "Note" that states "enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SCW System." Provide an explanation / technical justification for this additional guidance in the EPR Bases, Section B 3.7.9.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-121:

The Action "Note" for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Bases 3.7.9 "Safety Chilled Water (SCW) System" is provided because the Conditions and Required Actions have Completion Times that are greater than the Completion Times of the annotated supported systems. The Actions Section for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Bases 3.7.9 "Safety Chilled Water (SCW) System" contains the same information and level of detail provided in similar Actions Section in the Standard Technical Specifications for Westinghouse Plants NUREG-1431.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-122:

TS 3.6.6, Shield Building.

Revise SR 3.6.6.1, SR 3.6.6.3 and SR 3.6.6.4 to spell out "wg" as "water gauge", and to state that pressure is "equal to or more negative than" the limit; rather than ">" the limit. (Note that Bases use the phrase "negative pressure >" the limit.)

These changes will make TS 3.6.6 text consistent with the text content already established in the Bases B 3.6.6 and also in TS 3.7.12.

Response to Question 16-122:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.6.6.1, SR 3.6.6.3 and SR 3.6.6.4 will be revised to spell out "wg". The ">" symbol will be replaced with "equal to or more negative than".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.6 will be revised as described in the response and indicated on the enclosed markup.

Question 16-123:

TS 3.6.6, Shield Building.

Provide clarification for the difference in limit units used between the TS and the FSAR.

TS 3.6.6 specifies a value of "0.25 inches water gauge". EPR FSAR section 6.2.3.2.2.2 states the pressure to be maintained is less than or equal to "-0.09 psig." Identify the units will the installed instrumentation will measure. There should be consistency between the SR and FSAR measuring parameters so conversion factors do not introduce calculation errors.

This information will be used to verify adequacy and completeness of TS 3.6.6 surveillance requirements.

Response to Question 16-123:

U.S. EPR FSAR Tier 2, Chapter 6, Section 6.2.3.2.2 and Table 6.2.3-1 and Table 6.2.3-2 will be revised to state installed instruments will measure pressure in inches of water gauge.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 6 will be revised as described in the response and indicated on the enclosed markup.

Question 16-124:

TS 3.6.6, Shield Building.

Provide justification for not including a Surveillance Requirement to verify the structural integrity of the shield building. Revise TS 3.6.6 and the associated bases, as appropriate.

EPR GTS 3.6.6 does not have an equivalent to SR 3.6.8.3 in the Westinghouse STS for verification of the shield building structural integrity.

The information will be used to ensure all surveillance requirements are complete for EPR TS 3.6.6.

Response to Question 16-124:

An inspection of the Shield Building will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.6 and Bases.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.6 and Bases will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

uncontrolled release of radioactivity to the environment. The design description and performance criteria of the RSB are presented in Section 3.8.4.

The annulus ventilation system collects and filters airborne radioactive material that may leak from the primary containment by maintaining a subatmospheric pressure in the annulus.

6.2.3.2.2 Annulus Ventilation System

The AVS is designed to contain leakage from the primary containment by maintaining a subatmospheric pressure in the annulus. The AVS consists of three trains: one train is used during normal plant operation; two trains are used to mitigate potential accidents. AVS design and performance parameters are presented in Table 6.2.3-1.

Refer to Section 3.2 for the seismic and system quality group classification of the AVS.

6.2.3.2.2.1 AVS Normal Operation Train

The normal operation filtration train is shown in Figure 6.2.3-1. The full capacity normal operation filtration train is designed to maintain a subatmospheric pressure in the annulus, to maintain the annulus temperature above 45°F to prevent boron precipitation in the extra borating system piping, and to provide conditioned air in the annulus for personnel accessibility.

During normal operation, the conditioned air is drawn from the Nuclear Auxiliary Building ventilation supply shaft (See Section 9.4.3) through a fire damper, a motor-operated control damper, and two motor-operated isolation dampers. The supply air is distributed in the bottom of the annulus to four different locations. A subatmospheric pressure of less than or equal to ~~-0.03 psig~~ -0.8 inches water gauge is maintained in the annulus during normal operation by regulating the control damper with two redundant pressure sensors located in the annulus.

16-123

The exhaust air is drawn from the top of annulus by the Nuclear Auxiliary Building ventilation system exhaust fans through two motor-operated isolation dampers and a fire damper. The exhaust air is filtered by the Nuclear Auxiliary Building filtration trains and discharged through the vent stack.

The normal operation filtration train is in service during normal plant operation and plant shutdown conditions. The two accident trains are available as backup if the normal operation train is not able to maintain the subatmospheric pressure in the annulus.

The motor-operated air-tight dampers—located on the normal operation filtration train supply and exhaust ducts—isolate the secondary containment in case of a postulated accident. The redundant dampers in the supply and exhaust trains are powered by different electrical divisions backed by separate emergency diesel generators. The dampers can be operated automatically or manually from the main control room (MCR). In the event of a station blackout (SBO), these dampers are automatically closed by batteries.

The fire dampers on both supply and exhaust trains are located at the wall penetration between the Fuel Building and the annulus. These dampers are equipped with thermal sensors for automatic closing, and can be closed or re-opened remotely if not released by the thermal sensor.

6.2.3.2.2.2 AVS Accident Trains

The AVS accident filtration trains are shown on Figure 6.2.3-2. The filtration trains are engineered safety feature (ESF) filters and are used during postulated accidents to contain leakage from the primary containment by maintaining a subatmospheric pressure in the annulus. The exhaust air from the annulus is filtered before release to the environment via the vent stack.

There are two full capacity ESF trains, each consists of an air-tight motor-controlled damper, an electrical heater, a pre-filter, an upstream HEPA filter, an iodine absorber, a downstream HEPA filter, an air-tight motor controlled damper, a fan, and a back-draft damper. The filter system components are designed in accordance with Regulatory Guide 1.52, and are described in Section 6.5.1.

16-123

During a postulated accident, the ESF filtration trains collect the containment leakage from the annulus, remove airborne radioactivity through the filtration train, and release the filtered air to the vent stack. The AVS accident trains reduce the pressure in the annulus to at least ~~-0.09 psig~~ -0.25 inches water gauge or less and maintain the lower subatmospheric pressure. The system is capable of maintaining a uniform negative pressure throughout the secondary containment structure following the design basis loss of coolant accident (LOCA).

The exhaust air is monitored for radiation levels before release to the vent stack, as described in Section 12.3.4.

The two ESF trains are physically separated by being installed in separate rooms of the Fuel Building, which are also in separate fire areas. The two ESF trains are powered by different electrical divisions backed by separate emergency diesel generators.

6.2.3.2.2.3 System Operation

The normal operation filtration train is in service during normal plant operation, including cold shutdown and outages. During normal operation, the isolation dampers are in the open position and the annulus is continuously vented. The subatmospheric pressure inside the annulus is maintained by regulating the control damper located on

Table 6.2.3-1—Design and Performance of Annulus Ventilation System

Design Feature	Value
Maximum annulus pressure during normal operation [2]	≤ -0.8 inches water gauge~~0.03 psig~~~~[2]~~
Maximum annulus pressure during postulated accidents [2]	≤ -2.5 inches water gauge~~0.09 psig~~[2]
Minimum annulus temperature (all modes)	45°F
Maximum relative humidity at iodine filters (postulated accident)	70%
Design pressure	2.77 inches water gauge~~0.1 psig~~
Design temperature	212°F
Electrical heater power (each train)	6 kW
Minimum rated efficiency – Pre-filter	55-65%
Minimum rated efficiency – HEPA filters	99.95%
Minimum rated efficiency – Iodine adsorbers [1]	99% ~~[1]~~
Fan design air flow	60 – 1177 cfm

16-123

Note:

1. Laboratory test results for both elemental iodine and organic iodine, based on four (4) inch deep bed of carbon.
2. The subatmospheric pressure in the annulus will be equal to or lower than the value listed.

Table 6.2.3-2—Secondary Containment Response Analysis

Design Feature [3]	Value	
Annulus temperature	Initial	86.6°F
	After 24 hours	< 92°F
Annulus pressure	Start of drawdown	~~14.712 psia~~ 0.44 inches water gauge
16-123	At 305 seconds	~~14.686 psia~~ -0.25 inches water gauge
	After 565 seconds	~~14.609 psia~~ ≥ -2.5 inches water gauge
Annulus volume	Initial	706,299 ft^3
	After compression and at start of drawdown analysis	704,737 ft^3
Heat transfer coefficients [1, 2]	N/A [4]	
Conductive heat transfer [1]	N/A [4]	
Radiant heat transfer [1]	N/A [4]	
Compressive effect of primary containment [1]	Volume reduction of 1556 ft^3	
Secondary containment in-leakage assumed [1]	0.25% of containment free volume per day	
Secondary containment out-leakage assumed [1]	Zero leakage out of the secondary containment	
Heat loads generated within annulus [1]	Negligible	

Notes:

1. During postulated accident in primary containment.
2. Heat transfer calculated by methods provided in BTP 6-2.
3. Secondary containment response analysis based on worst single failure.
4. An infinite heat transfer coefficient was assumed such that the surface temperature in contact with primary containment is at the design maximum value from time zero.

Next File

Methodology

The Type A test is conducted in accordance with ANSI N45.4, the Total Time and Point-to-Point methods, or ANSI/ANS-56.8, the Mass Point Method. The duration of testing after the stabilization period is established consistent with these standards. Changes in containment air mass during this time are calculated using periodic measurements of containment pressure, dry bulb temperature, and dew point temperatures (i.e., water vapor pressure).

Accuracy of the Type A test results is then verified by a supplemental verification test. The supplemental verification test is performed using the methodology for this purpose described by ANSI N45.4.

Type A tests use temporary air compressors, which are installed and connected to the permanent system piping. The number and capacity of the compressors is sufficient to pressurize the primary containment to test pressure. The compressors include air coolers, moisture separators, and air dryers to reduce the moisture content of the air entering the primary containment. Temperature and humidity sensors are installed inside containment for Type A testing. Data acquisition hardware and instrumentation is available outside containment. Instrumentation that is not required during normal plant operation may be installed temporarily for the Type A test.

The maximum allowable containment leakage rate denoted L_a, is not to exceed 0.25% of containment air ~~mass~~weight per day at P_a. The conservative value for P_a, 55 psig, is greater than the calculated peak internal pressure associated with the design basis accident (DBA). 16-117

As soon as practical after completion of a Type A test that identified leakage, and prior to initiation of containment inspection for the subsequent Type A test, repairs or adjustments are made to components that exceeded individual leakage limits.

Preservice Test

Preoperational Type A tests are conducted at peak pressure in accordance with 10 CFR 50, Appendix J, as follows:

- Peak pressure tests—A test is performed at pressure P_a to measure the leakage rate L_{am}.
- Peak pressure acceptance criteria—The leakage rate L_{am} is less than 0.75 L_a at pressure P_a.
- If the Type A test fails to meet this criterion, a retest is conducted in accordance with the requirements of 10 CFR 50, Appendix J.

Table 6.3-4—IRWST Design Parameters

16-95

Parameter	Value
IRWST	
Design pressure	75 psig
Design temperature	320°F
Minimum operating temperature	59°F
Maximum operating temperature	122°F
Minimum volume	66,886 ft^3 (500,342 gallons)
Maximum volume	70,010 ft^3 (523,703 gallons)
Minimum boron enrichment	37% of B-10
Minimum boron concentration	1700 ppm
Maximum boron concentration	1900 ppm
SIS Sumps Screen Filters	
Number	4
Material	Austenitic stainless steel
Design pressure	75 psig
Design temperature	320°F
Opening size	0.08 x 0.08 in
Diameter of wire	0.03 in
Total screen area (approx.)	753 ft^2
SIS Vortex Suppressors	
Number	4
Material	Austenitic stainless steel
Design pressure	75 psig
Design temperature	320°F
Liner	
Material for fluid wetted parts	Austenitic stainless steel
Liner plate thickness	
Wall	0.39 in
Bottom	0.39 in
Ceiling	0.39 in
Area	
Wall	6350.7 ft^2
Bottom	5866.3 ft^2
Ceiling	1829.9 ft^2
Sump	484.4 ft^2
Water depth (approximate)	12.3 ft

The auxiliary boiler system consists of feedwater supply equipment, deaerator, sampling system, water chemistry control equipment and automatic control equipment. Safety relief valves are in accordance with the ASME Boiler and Pressure Vessel (BPV) Code, Section VIII (Reference 16) to protect the system from overpressuration. The auxiliary boiler system does not perform any safety-related functions.

10.3.2.2 Component Description

Table 3.2-1 provides the quality group and seismic design classification of components and equipment in the MSSS. Section 3.2 also describes how the guidance in RG 1.26 is implemented for the U.S. EPR. The main steam lines, from the SGs up to and including the fixed restraint downstream of the MSIVs, are designed and constructed in accordance with Quality Group B and Seismic Category I. The remaining piping out to the TG stop valve and second stage reheaters meets ASME Power Piping Code B31.1 (Reference 2). Data related to containment isolation for MSSS valves are listed in U.S. EPR FSAR Tier 2, Table 6.2.4-1—Containment Isolation Valve and Actuator Data.

Main Steam Safety Valves

Each main steam line has two Reference 1 safety valves, located upstream of the main steam isolation valve (MSIV). The main steam safety valves (MSSV), along with the main steam relief trains (MSRT), provide overpressure protection of the main steam piping and SGs. The safety valves discharge to the atmosphere via directly connected vent stacks. Low-point drains in the vent stacks route any accumulated water to the TB drains.

16-86

~~MSSV setpoint and capacity are such that with consideration of reactor trip (RT), the MSSVs alone prevent system pressure from rising above 110 percent of design value upon full loss of load.~~

Table 10.3-2—Design Data for Main Steam Safety Valves, provides design data for the MSSVs.

Main Steam Relief Trains

Each main steam line has one MSRT located upstream of its MSIV. Each MSRT consists of a Reference 1 normally closed, fast opening MSRIV and a downstream, normally open MSRCV. The MSRIVs are designed in accordance with ASME BPV Code, Section III, Division 1, Subsection NC including Article NC-7000 (Reference 3).

The MSRTs are part of the SG secondary side overpressure protection. MSRT setpoint and capacity are such that with consideration of RT, the MSRTs alone prevent system pressure from increasing above 110 percent of design pressure upon full loss of load.

Table 3.3.2-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED NUMBER OF DIVISIONS
1. Cold Leg Temperature (Wide Range)	1 per loop
2. Containment Isolation Valve Position Indication	2[(a)(b)]
3. Containment Pressure	2
4. Emergency Feedwater Storage Pool Level	1 per pool
5. Emergency Feedwater System Flow	1 per loop
6. Extra Borating~~on~~ System Flow	2
7. Hot Leg Injection Flow	1 per loop
8. Hot Leg Pressure (Wide Range)	1 per loop
9. Hot Leg Temperature (Wide Range)	1 per loop
10. In-~~c~~Containment Refueling Water Storage Tank Level	2
11. Incore Temperature	2 per quadrant
12. Power Range Monitors	2
13. Pressurizer Level	2
14. Radiation Monitor - Containment High Range	2
15. Radiation Monitor - Main Steam Line Activity	1 per line
16. Source Range Monitors	2
17. Steam Generator Level (Wide Range)	2 per SG
18. Steam Generator Pressure	2 per SG

16-100

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication division is required for penetration flow paths with only one installed control room indication division.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average coolant temperature (T_{avg}) shall be ≥ 568°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

16-73.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 568°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two RCS loops shall be OPERABLE and one RCS loop shall be in operation.

OR

Three Residual Heat Removal (RHR) loops shall be OPERABLE and two RHR loops shall be in operation.

--NOTE--
All reactor coolant pumps and LHSI pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop or one or two required RHR loops inoperable.	A.1 Initiate action to restore required loops to ~~OPERABLE~~ OPERABL status. 16-75 AND	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
16-73.6	A.2 -------------NOTE------------- Only required if two RHR loops are OPERABLE. ---------------------------------- ~~A.2~~ Be in MODE 5.	 24 hours
B. Two RCS or three RHR~~more~~ required loops inoperable. OR Required loop(s) not in operation. 16-75	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RCS loop or two RHR loops to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required RCS or RHR loops are in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 20% for required RCS loops.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
	OR ~~OR~~ 16-73.13	
	A.2.2 Restore RCS PIV 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 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----------------------------NOTES---------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when operating in the RHR mode. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. --- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5.0 gpm at an RCS pressure of ≥ 2215 psig and ≤ 2255 psig. 16-71	In accordance with the Inservice Testing Program, and 24 months AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (SG) Tube Integrity

LCO 3.4.16 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube ~~plugging~~repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND 16-73.15 U-line B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	Verify isotopic concentration of B~~^{10}~~-10 in each accumulator is ≥ 37%.	24 months

16-109

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Four ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MHSI train inoperable.	A.1 Restore MHSI train to OPERABLE status.	120 days
B. One LHSI train inoperable.	B.1 Open ECCS cold leg cross connections. AND B.2 Restore LHSI train to OPERABLE status.	72 hours 16-102 ~~60~~ 120 days
C. Two ~~MHSI~~ ECCS trains inoperable. ~~OR~~ ~~Two LHSI trains inoperable~~	C.1 Restore one inoperable train to OPERABLE status.	72 hours

16-93

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 Two Medium Head Safety Injection (MHSI) trains shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MHSI train inoperable.	A.1 Restore required MHSI train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two required MHSI trains inoperable.	B.1 Be in ~~Mode~~ MODE 5. 16-100	12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 In-Containment Refueling Water Storage Tank (IRWST)

LCO 3.5.4 The IRWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. IRWST temperature, water volume, boron concentration, or enrichment not within limits.	A.1 Restore IRWST temperature, water volume, boron concentration, and enrichment to within limits.	8 hours
B. IRWST inoperable for reasons other than Condition A.	B.1 Restore IRWST to OPERABLE status.	1 hour
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

16-95

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify IRWST borated water temperature is ≥ 59°F and ≤ 122°F.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.4.2	Verify IRWST borated water volume is ≥ 500,342 gallons and ≤ 523,703 gallons.	7 days
SR 3.5.4.3	Verify IRWST boron concentration is ≥ 1700 ppm and ≤ 1900 ppm enriched boron.	7 days
SR 3.5.4.4	Verify isotopic concentration of $B^{\cancel{10}}$-10 in the IRWST is ≥ 37%.	24 months

16-109

16-100

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Extra Borati~~on~~ng System (EBS)

LCO 3.5.5 Two EBS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both EBS tanks inoperable due to boron concentration or enrichment not within limits.	A.1 Restore boron concentration and enrichment to within limits.	72 hours
B. One EBS train inoperable for reasons other than Condition A.	B.1 Restore EBS train to OPERABLE status.	7 days
C. Two EBS trains inoperable for reasons other than Condition A.	C.1 Restore one EBS train to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	~~12~~ 6 hours 36 hours

16-98

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	Verify each EBS tank borated water temperature and EBS pump room is ≥ 68°F. 16-99	24 hours
SR 3.5.5.2	Verify total EBS tank borated water volume is ≥ 2345 ft^3.	7 days
SR 3.5.5.3	Verify each EBS tank boron concentration is ≥ 7,000 ppm and ≤ 7,300 ppm enriched boron.	31 days AND Once within 24 hours after water or boron is added to tank AND Once within 24 hours after tank temperature is restored to within limit
SR 3.5.5.4	Verify each EBS train manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.5.5.5	Verify each EBS pump develops a flow rate ≥ 49.0 gpm and ≤ 55.4 gpm.	In accordance with the Inservice Testing Program
SR 3.5.5.6	Verify isotopic concentration of B~~10~~-10 in each EBS tank is ≥ 37%.	24 months

16-109

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.5.7	Verify flow through one EBS train from the pump into the RCS.	24 months ~~on a STAGGERED TEST BASIS~~
SR 3.5.5.8	Verify each EBS train power operated outboard containment isolation valve can be manually aligned to the open position from the main control room (MCR) after automatic closure of the outboard containment isolation valve by the Containment Isolation Signal.	24 months

16-101

16-96

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 path(s), except for the Full Flow Purge 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 systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve 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 or more containment isolation valves. 16-116 --------------------------- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D.	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

16-116

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each Full Flow Purge valve is sealed closed, except for one full flow purge valve in a penetration flow path while in Condition D of this LCO.	31 days
SR 3.6.3.2	Verify each Low Flow Purge valve is closed, except when the Low Flow 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 controls. -- Verify each containment isolation manual valve and blind flange that is located outside 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.	31 days
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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.6 16-116	Perform leakage rate testing for Full Flow and Low Flow 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.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.6 Shield Building

LCO 3.6.6 The Shield Building shall be OPERABLE.

---NOTE---
The Shield Building envelope may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shield building inoperable.	A.1 Restore shield building to OPERABLE status.	24 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
	16-122	
SR 3.6.6.1	Verify annulus negative pressure is ≥ 0.25 inches ~~wg~~water gauge.	12 hours
SR 3.6.6.2	Verify each shield building access door is closed, except when the access opening is being used for entry and exit.	31 days

SURVEILLANCE REQUIREMENTS

16-122

	SURVEILLANCE	FREQUENCY
SR 3.6.6.3	Verify the annulus can be drawn down to a pressure ~~≥~~equal to or more negative than 0.25 inches ~~wg~~ water gauge using one Annulus Ventilation System (AVS) train in ≤ 305 seconds after a start signal.	24 months on a STAGGERED TEST BASIS for each AVS train
SR 3.6.6.4	Verify the annulus can be maintained at a pressure ~~≥~~ equal to or more negative than 0.25 inches ~~wg~~ ater gauge by one AVS train at a flow rate of ≤ 1295 cfm.	24 months on a STAGGERED TEST BASIS for each AVS train
SR 3.6.6.5	Verify Shield Building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the Shield Building.	During shutdown for SR 3.6.1.1 Type A tests

16-124

3.6 CONTAINMENT SYSTEMS

3.6.7 Annulus Ventilation System (AVS)

LCO 3.6.7 Two AVS accident filtration trains shall be OPERABLE.

16-112 & 16-113

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AVS accident filtration train inoperable.	A.1 Restore AVS accident filtration train to OPERABLE status.	7 days
B. One or more normal operation train isolation dampers inoperable.	B.1 Restore isolation damper to OPERABLE status.	24 hours
~~B~~C. Required Action and associated Completion Time not met.	~~B~~C.1 Be in MODE 3. AND ~~B~~C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Operate each AVS accident filtration train for ≥ 15 minutes with heaters operating.	31 days
SR 3.6.7.2	Perform required AVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.7.3	Verify each AVS accident filtration train actuates on an actual or simulated actuation signal.	24 months
SR 3.6.7.4	Verify that the normal operation train motor operated isolation dampers close on an actual or simulated isolation signal.	24 months

16-112 & 16-113

16-112

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Two MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSSV inoperable.	A.1 Verify associated Main Steam Relief Train is OPERABLE. AND	Immediately 16-86
	A.2 Restore MSSV to OPERABLE status.	~~30 days~~72 hours
~~B. Two MSSVs inoperable.~~	~~B.1 Restore one MSSV to OPERABLE status.~~	~~7 days~~
~~C~~B. Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR ~~Three~~ Two or more MSSVs inoperable.	~~C~~B.1 Be in MODE 3. AND ~~C~~B.2 Be in MODE 4.	6 hours 12 hours

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater (MFW) Valves

LCO 3.7.3 Four MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), MFW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs) shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when ~~all~~ MFW~~s~~ valves are closed and deactivated.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more full load flow paths with one MFW valve inoperable.	A.1 Restore MFW valve to OPERABLE status.	7 days
B. One or more full load flow paths with two MFW valves inoperable.	B.1 ~~Restore one MFW valve to~~Close or isolate the affected MFW full load flow path ~~OPERABLE status~~.	72 hours
	B.2 Verify the affected MFW full load flow path is closed or isolated.	Once per 7 days
C. One or more full load flow paths with three MFW valves inoperable.	C.1 ~~Restore one MFW valve to OPERABLE status~~Isolate the affected flow path.	8 hours

16-92

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration (CREF)

LCO 3.7.10 Two CREF trains shall be OPERABLE.

---NOTE---
The control room envelope (CRE) may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF train inoperable for reasons other than Condition B.	A.1 Restore CREF train to OPERABLE status.	7 days
B. ~~Two~~ One or more CREF trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4	B.1 Initiate action to implement mitigating actions.	Immediately
	AND	
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological, ~~[[~~chemical,~~]]~~ and smoke hazards will not exceed limits.	24 hours
	16-63	
	AND	
	B.3 Restore CRE boundary to OPERABLE status.	~~90~~ 60 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CREF trains inoperablein MODE 5 or 6, or during movement of irradiated fuel assemblies. OR One or more CREF trains inoperable due to an inoperable CRE boundary in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies. 16-63	Immediately
F. Two CREF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

16-109

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Boron Concentration and Enrichment

-- REVIEWER'S NOTE --
The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.

LCO 3.7.15 The spent fuel storage pool boron concentration shall be ≥ ~~1291~~ [1334] ppm and boron enrichment shall be ≥ 37%.

16-109

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel storage pool boron concentration is within limit.	7 days
SR 3.7.15.2	Verify the isotopic concentration of B-10 in the spent fuel storage pool is within limit.	24 months

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions.~~, except the introduction of coolant into the Reactor Coolant System (RCS).~~	Immediately
16-107	AND	
	A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration."	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

16-104

3.9 REFUELING OPERATIONS

3.9.3 Decay Time

LCO 3.9.3 The reactor shall be subcritical for ≥ 34 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 34 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Verify that the reactor has been subcritical for ≥ 34 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.3	Initiate action to satisfy the requirements of the LCO.	Immediately
	AND		
	A.4	Close equipment hatch and secure ~~with four bolts~~.	4 hours
		16-83	
	AND		
	A.5	Close one door in each air lock.	4 hours
	AND		
	A.6	Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Ventilation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loops in operation.	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration."	Immediately
	AND		
	B.2	Initiate action to restore required RHR loop to operation.	Immediately
	AND		
16-83	B.3	Close and secure equipment hatch ~~and secure with four bolts~~.	4 hours
	AND		
	B.4	Close one door in each air lock.	4 hours
	AND		
	B.5	Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Ventilation System	4 hours

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, extra ~~boration~~ borating system (EBS), or the In-containment Refueling Water Storage Tank (IRWST). The operator should borate with the best source available for the plant conditions.

16-100

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.6.2

Verification of the control bank insertion, sequence, and overlap limits is required to detect control banks that may be approaching the insertion, sequence and overlap limits. During normal operation it is unlikely that this should be a problem since normally, very little rod motion occurs in 12 hours and automatic rod motion is controlled by the Reactor Control, Surveillance and Limitation System (RCSL). The operator is responsible for monitoring each RCSL automatic movement of the control banks and to take manual actions if necessary to maintain insertion, sequence, and overlap limits. A Completion Time of 12 hours is adequate to ensure that the control banks are maintained within insertion, sequence, and overlap limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28~~, August, 2007~~.

2. 10 CFR 50.46~~, August, 2007~~.

3. NUREG-0800, Section 4.2, Appendix B.

16-73.1

4. FSAR Chapter 15.

BASES

ACTIONS (continued)

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where RCCA position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the RCCA positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the analog RCCA position indication agrees with the digital RCCA position within 8 steps ensures that the analog RCCA position indication is operating correctly. Since the analog RCCA position indication does not display the actual RCCA position below 10 steps, only points within the indicated ranges are required in comparison.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 26, and GDC 28~~, August, 2007~~.

~~2. 10 CFR 50.46, August, 2007.~~

~~3~~2. FSAR Chapter 15.

16-73.1

BASES

ACTIONS (continued)

B.1 and B.2

If the VCT and letdown isolation valves cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a configuration in which the LCO does not apply. As an alternative (Required Actions B.1 and B.2), the VCT and letdown return line line must be closed and secured within 8 hours to isolate the unborated water sources. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to return the isolation valves to an OPERABLE condition in an orderly manner. The 31 day frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that make valve opening an unlikely possibility.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Periodic surveillance testing of VCT and letdown isolation valves is required by the ASME Code. This verifies that the measured performance, receipt of isolation signal to full closure, is within an acceptable tolerance of the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.1.8.2

This periodic surveillance is performed on the VCT and letdown isolation valves to verify that the actuation signal causes the appropriate valves to move to their correct position within the allowable design basis response time.

The 24 month frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The Frequency is acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Protection System testing.

REFERENCES

1. FSAR ~~Chapter 15~~Section 15.4.6.

16-73.1

2. 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

BASES

BACKGROUND (continued)

Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the LTSP and the methodology for calculating the as-left and as-found tolerances, for the phrase "a document controlled under 10 CFR 50.59" in the specifications.

The LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the LTSP accounts for uncertainties in setting the device (e.g., CALIBRATION), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments (Ref. 6)). In this manner, the LTSP ensures that SLs are not exceeded. As such, the LTSP meets the definition of a SL-LSSS (Ref. 1). The LTSPs are determined as part of the safety analysis (Ref. 5).

16-73.1

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." However, use of the LTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protective device setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the LTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the LTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the trip setpoint to account for further drift during the next surveillance interval.

However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This LTSP specified in Table 3.3.1-2 is the least conservative value of the as-found setpoint that a channel can have during testing such that a channel is OPERABLE if the trip setpoint is found conservative with respect to the Allowable Value during a SENSOR

BASES

LCO (continued)

5. Emergency Feedwater System Flow

Emergency Feedwater flow indication is required when throttling feedwater flow to the steam generators. Control of flow is required to control the rate of steam generator heat removal to maintain Reactor Coolant temperature profiles within limits for cooldown.

16-100

6. Extra ~~Boration~~ Borating System Flow

The Extra ~~Boration~~ Borating System flow provides verification that the appropriate system alignment has been completed. The negative reactivity additions performed by this system require verfication of correct system operation.

7. Hot Leg Injection Flow

Hot Leg Injection flow provides verification that the appropriate system alignment has been completed. Hot leg injection is required to prevent the buildup of sufficient boron concentration in the core coolant channels to impede long term core cooling.

8. Hot Leg Pressure (Wide Range)

RCS pressure is required to monitor reactor coolant integrity and assess core cooling. RCS pressure and either RCS hot leg or incore temperature is used to determine subcooling margin if the calcuation is not available.

9. Hot Leg Temperature (Wide Range)

Hot Leg Temperature is required to monitor core cooling, to verify natural circulation, and to verify primary to secondary loop coupling along with steam generator pressure. Hot Leg temperature and RCS pressure are used to determine loop subcooling margin if the calculation is not available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 24 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 24 months reflects the importance of verifying flow and has been shown by operating experience to be acceptable.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. FSAR Chapter 15.

16-73.1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 568°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

1. FSAR Chapter 15.

16-73.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACK~~C~~GROUND

16-73.3

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

The Pressure Temperature Limit Report (PTLR) contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

BASES

REFERENCES	1. FSAR Chapter 15.

16-73.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

16-73.5

16-73.4

Whenever the Control Rod Drive Control System (CRDCS) is capable of rod withdrawal an inadvertent rod withdrawal from subcritical is possible, resulting in a power excursion. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible ~~ir~~regardless of CRDCS capabilities. Such a transient could be caused by the mechanical failure of a Control Rod Drive Mechanism (CRDM).

Therefore, in MODE 3 with the CRDCS capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires four RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the CRDCS is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None~~Chapter 15~~.

16-73.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND	In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the Low Head Safety Injection (LHSI) heat exchangers connected to the Residual Heat Removal (RHR) System. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.
16-75	In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or two RHR loops for adequate decay heat removal ~~and transport~~. One RCP or RHR loop provides adequate forced flow for transport. ~~The flow provided by one RCP loop or two RHR loops is adequate for decay heat removal.~~ The other intent of this LCO is to require that additional paths be available to provide redundancy for decay heat removal.
APPLICABLE SAFETY ANALYSES	In MODE 4, RCS loop circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation. RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	The purpose of this LCO is to require that at least two RCS loops or three RHR loops are OPERABLE in MODE 4, and that one of the RCS loops or two of the RHR loops are in operation. Any one RCS loop or two RHR loops in operation will provide enough flow to remove the decay heat from the core and allow for cooldown. Any one RCS or RHR loop in operation will provide sufficient coolant transport for boron mixing. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RCS or two loops of RHR provides sufficient circulation for ~~these purposes~~decay heat removal. One RCS or RHR loop provides boron mixing. However, additional loops consisting of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

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, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1 ~~and A.2~~

If one required RCS or RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or third RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and two RHR loops are OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only two RHR loops OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of one of the two remaining RHR loops, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if two RHR loops are OPERABLE. With one RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of an RHR loop, rather than a cooldown of extended duration.

16-64 & 16-80

BASES

ACTIONS (continued)

B.1 and B.2

16-80

If two ~~or more~~RCS or three RHR required loops are inoperable or the required loop(s) are not in operation, except during conditions permitted by the Note in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1~~, "SHUTDOWN MARGIN (SDM)"~~ must be suspended and action to restore one RCS or two RHR loops to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1~~, "SHUTDOWN MARGIN (SDM)"~~ is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until the required loop(s) are restored to OPERABLE status or operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that the required RCS or RHR 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 Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 20%. If the SG secondary side narrow range water level is < 20%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

16-78

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or LHSI ~~/ RHR~~ pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. 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.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation. This is acceptable because the proper breaker alignment and power availability are ensured if a pump is operating.

REFERENCES None.

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-78

SR 3.4.7.3

Verification that each required ~~RHR~~ LHSI pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required ~~RHR~~ LHSI pump. If secondary side water level is ≥ 20% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation. This is acceptable because proper breaker alignment and power availability are ensured if a pump is operating.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

16-78

In MODE 5 with loops not filled, only ~~RHR~~ LHSI pumps can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one ~~RHR~~ LHSI pump 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 the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (Loops Not Filled) satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running LHSI pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

An OPERABLE RHR loop is comprised of an OPERABLE LHSI pump capable of providing forced flow to an OPERABLE LHSI heat exchanger. LHSI pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

BASES

BACKGROUND (continued)

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSRV with water overflow. The CVCS charging and auxiliary spray valves isolate on an increasing pressurizer level to perform this function. The CVCS charging isolation is normally open during operation for pressurizer level control while the CVCS auxiliary spray valve is opened during plant cooldown to reduce pressurizer temperature and provide pressure control when the RCS normal sprays are not functional (RCPs removed from service).

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

16-73.7

Safety analyses presented in FSAR Chapter 15 (Ref. 1) do not take credit for emergency supply pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSRV with water overflow. The closure of the CVCS charging and auxiliary spray valves perform this function.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii). Although the emergency supply heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1240 cubic feet, which is equivalent to 75%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated emergency supply pressurizer heaters are verified to be at their design rating. This SR may be verified by energizing the heaters and measuring circuit current. The frequency of 92 days is considered adequate to detect heater degradation and has shown by operating experience to be acceptable.

SR 3.4.9.3

These Surveillances demonstrate that each automatic valve used to isolate the pressurizer actuates to the required position on an actual or simulated PS signal. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 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 Protection System testing, and equipment performance is monitored as part of the Inservice Testing Program.

REFERENCES

16-73.1

1. FSAR Chapter 15.

2. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Relief Valves (PSRVs)

BASES

BACKGROUND

The p~~P~~ressurizer s~~S~~afety r~~R~~elief v~~V~~alves (PSRVs), in conjunction with the ~~Reactor~~ Protection System ~~(RPS)~~ provide overpressure protection for the RCS. The PSRVs are pilot operated relief valves. Each relief valve has two types of pilot valves. The spring operated pilot valves are used to detect and provide the opening setpoint for the relief valve. The second type of pilot valve is the solenoid type. The solenoid pilot valves are used only for low temperature overpressure protection.

16-73.9

Each PSRV has two spring operated pilot valves that are arranged in a parallel configuration; only one pilot valve has to be operated to allow the PSRV to open. Both pilot valves are tested and adjusted to the proper relief valve setpoint. Normally one pilot valve is in operation (un-isolated) and the other is manually isolated and considered a standby device. In the event the in service pilot valve is suspected of being faulty the standby pilot can be placed in service, and the other valve isolated for troubleshooting. If it is determined that both pilot valves are faulty the PSRV would be declared inoperable. The PSRVs are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2803 psia~~,~~ (2788 psig) which is 110% of the design pressure.

16-73.8

Because each of the three safety relief valves are totally enclosed and self actuating they are considered independent components. The relief capacity for each valve, 661,400 lb_m/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the PSRVs is directed to the p~~P~~ressurizer r~~R~~elief t~~T~~ank. This discharge flow is indicated by an increase in temperature downstream of the PSRVs or increase in the ~~P~~pressurizer r~~R~~elief t~~T~~ank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures less than or equal to LTOP arming temperature specified in the PTLR, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by meeting the requirements of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 when any RCS cold leg temperature~~s~~ is~~are~~ less than or equal to the LTOP arming temperature specified in the PTLR~~.~~ or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head removed.

16-73.10

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the PSRVs at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one PSRV inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the Pressurizer Relief System. An inoperable safety relief valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

16-73.11

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more PSRVs are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures less than or equal to the LTOP arming temperature specified in the PTLR 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 and without challenging plant systems. With any RCS cold leg temperatures less than or equal to the LTOP arming temperature specified in the PTLR overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large Pressurizer insurges, and thereby removes the need for overpressure protection by three PSRVs.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. PSRVs are to be tested in accordance with the requirements the ASME Code (Ref. 3), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The PSRV setpoint is ± 2 % for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. FSAR ~~Chapter 15~~Table 15.0-10.

16-73.1 & 16-73.6

3. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants, 2004.

BASES

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR Chapter 15.

16-73.1

5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME Code for Operation and Maintenance of Nuclear Power Plants.

16-73.1

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses (Ref. 3) 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 that primary to secondary LEAKAGE is 0.125 gallon per minute (gpm) per steam generator (SG) or increases to 0.125 gpm per SG as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through ~~each~~any one SG to less than or equal to 150 gallons per day is lower than the primary to secondary leakage value used in the safety analysis.

16-73.12

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from variety of accidents (such as a main steam line break, steam generator tube rupture, rod ejection accident, RCS pump locked rotor, etc.). The basic radiological acceptance criteria associated with the alternative source term (AST) methodology are found in 10 CFR 50.34(a)(1) for the offsite receptors, with a limit of 25 rem total effective dose equivalent (TEDE). 10 CFR Part 50, Appendix A, GDC 19 as incorporated by reference in 10 CFR 52.47(a)(1), includes the criteria for control room personnel (5 rem TEDE). These criteria, however, are used for evaluating potential reactor accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, the acceptance criteria for the offsite receptors are more stringent, while the criteria for the control room operators remains the same.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(~~d~~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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG, in which case the LEAKAGE rate limit of 150 gallons per day would still apply.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45 May 1973.
3. FSAR Chapter 15.
4. NEI 97-06, Steam Generator Program Guidelines.
5. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

16-73.1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2, SR 3.4.14.3, and SR 3.4.14.4

These SRs require the performance of a 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 24 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. FSAR ~~Chapter 5~~Section 5.2.5.

16-73.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

The maximum ~~Total Effective~~ dose that~~equivalent (TEDE)~~ an individual at the ~~E~~exclusion ~~A~~area ~~B~~boundary can receive for 2 hours following an accident, or at the ~~L~~low ~~P~~population ~~Z~~zone outer boundary for the radiological release duration, is specified in ~~10 CFR 50.67~~10CFR100.21 (Ref. 1). Doses to the ~~C~~control ~~R~~room operators must be limited per GDC 19. The limits on specific activity ensure that the doses are appropriately limited during analyzed transients and accidents.

16-81

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a main steam line break (MSLB) or a steam generator tube rupture (SGTR) ~~or a main steam line break (MSLB)~~ accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and ~~C~~control ~~R~~room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure~~s~~ that the resulting offsite and ~~C~~control ~~R~~room doses meet the appropriate S~~tandard~~ R~~eview~~ P~~lan~~ acceptance criteria following an MSLB~~SGTR~~ or an ~~MSLB~~ SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit. The MSLB safety analysis (Ref. 4) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit and an existing reactor coolant system generator (SG) tube leakage rate of 0.125 gpm in the affected steam generator and 0.375 gpm combined in the unaffected steam generators. The safety analysis for both accidents assumes the specific activity of the secondary coolant at its limit of 0.1 µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

The analysis for the MSLB and SGTR accidents establish the acceptance ~~to~~ limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

16-81

BASES

16-81

ACTIONS (continued)

B.1

With the DOSE EQUIVALENT XE-133 ~~in excess of the allowed~~greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limits within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of an MSLB or SGTR occurring during this time period.

A NOTE permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which ~~is limiting~~ due to exceeding this limit, and the ability to restore transient~~-~~ specific activity excursions while the plant remains at, or proceeds to~~,~~ power operation.

16-73.14

16-81

C.1 and C.2

If any Required Action and the associated Completion Time of Condition A or B is not met or if the DOSE EQUIVALENT I-131 is > ~~60.0~~1.0 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in noble gas specific activity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

16-81

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.15.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR ~~50.67~~100.21. 16-81

2. ~~Standard Review Plan (SRP)~~NUREG-0800 Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

16-73.1

3. FSAR Section 15.0.3.7.

4. FSAR Section 15.0.3.6.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

16-73.16

The Steam Generator Program defines the Frequency of SR 3.4.~~17~~16.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.16.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~plugging~~repair criteria is removed from service by plugging. The tube ~~plugging~~repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube ~~plugging~~repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the ~~plugging~~repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, boron isotopic inventory, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

16-84

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator ~~motor operated~~motor-operated isolation valves are closed to isolate the accumulators from the RCS. During RCS cooldown, ~~two~~ one accumulator~~s (Trains 3 and 4) are~~is depressurized to approximately 304 psig and reconnected to the RCS to prevent Reactor Coolant Pump (RCP) seal injection damage in the event of an inadvertent RCS depressurization when the pressurizer is in a water solid state. Once all RCPs are stopped, the ~~Train 3 and 4~~ accumulator~~s~~ is ~~are~~ again isolated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the in-containment refueling water storage tank (IRWST), because the water contained in the IRWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator ~~motor operated~~motor-operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

~~This~~ The SR is modified by a Note that requires electrical power~~allows power to be~~ supplied to the motor~~-~~-operated isolation valves to be verified de-energized when RCS pressure is ≥~~<~~ 2000 psig.~~, thus allowing~~ This allows operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

SR 3.5.1.6

The boron used in the accumulators is enriched to ≥~~>~~ 37% in the B-10 isotope. Verification every 24 months that the B-10 enrichment is ≥~~>~~ 37% ensures that the B-10 concentration assumed in the accident analysis is available. Since B-10 in the accumulators is not exposed to a significant neutron field, 24 months is considered conservative.

REFERENCES

1. FSAR ~~Chapter 15~~Section 15.6.
2. 10 CFR 50.46.
3. FSAR ~~Chapter 6~~Section 6.2.
4. NUREG-1366, February 1990.

16-73.1

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and hot leg recirculation. In the injection phase, water is taken from the in-containment refueling water storage tank (IRWST) and injected into the Reactor Coolant System (RCS) through the cold legs. After approximately ~~24~~20 hours, the LHSI flow is shifted to ~~the~~ hot leg recirculation ~~phase~~ to ~~provide a~~ backflush the loops, ~~which would~~ reduce the boiling in the top of the core, and recapture any ~~resulting~~ boron precipitation.

16-93

~~The ECCS consists of two separate subsystems: Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI). Each subsystem consists of four redundant, 100% capacity trains. The ECCS accumulators and the IRWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.~~ Each ECCS train consists of two separate subsystems: Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI). Each ECCS flow path consists of piping, valves, heat exchangers, and pumps that provide a means to transfer the water from IRWST to one RCS loop cold leg. Four separate suction supply lines supply water from the IRWST to the ECCS pumps. Each ECCS train is capable of providing 100 percent of the required flow to mitigate the design basis accident (DBA). With the ECCS cross-connects closed, each of the four trains is independent and injects into a single RCS cold leg.

BASES

BACKGROUND (continued)

If it is necessary to remove one LHSI train from service, an isolatable ECCS cross-connect ensures LHSI delivery in the event of a cold leg break. Whenever the cross-connects are opened, the isolation valve electrical breakers are racked-out to avoid single failure. Otherwise, both ECCS cross-connects are isolated to maintain train separation.

The ECCS accumulators and the IRWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

Cross-connect capability is provided between Trains 1 and 2 and between Trains 3 and 4 for the LHSI pumps. Cross-connect capability allows one SI train to be removed from service for maintenance and provides alternate injection paths for the LHSI trains that remain in service. This configuration mitigates the effect of potential degraded injection flow due to steam entrainment during a loss of coolant event. At higher pressures when the MHSI pumps provide the injection flow, steam entrainment is not a concern; therefore no cross-connects are required between the MHSI trains.

16-93

Each ECCS flow path consists of piping, valves, heat exchangers, and pumps such that water from the IRWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the MHSI pumps, the LHSI pumps, and heat exchangers. Each of the two subsystems (MHSI and LHSI) consists of four 100% capacity trains that are independent and redundant such that each train is capable of supplying 100% of the flow required to mitigate the accident consequences.

BASES

BACKGROUND (continued)

~~Four separate suction supply lines supply water from the IRWST to the ECCS pumps. Each of the four trains is independent and injects into a single RCS cold leg. If it is necessary to remove one LHSI train from service, an isolatable ECCS cross-connect ensures LHSI delivery in the event of a cold-leg break. Whenever the cross-connects are opened, the isolation valve's electrical breakers are racked-out to avoid single failure. Otherwise, both ECCS cross-connects are isolated to maintain train separation.~~

16-93

For LOCAs that are too small to depressurize the RCS below the shutoff head of the MHSI pumps, the secondary side is cooled down to approximately 870 psia at a rate of approximately 180°F/hr by means of the relief valves to ensure adequate injection from the MHSI system.

Due to the large miniflow lines, it is not necessary to limit the number of MHSI or LHSI pumps in service during low temperature conditions in the RCS. Refer to the Bases for LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for the basis of low RCS temperature operation.

The ECCS subsystems are actuated upon receipt of a Protection System (PS) signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the IRWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "In-Containment Refueling Water Storage Tank (IRWST)," provide the Cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

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 post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The MHSI and LHSI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the MHSI pumps. The SGTR and MSLB events also credit the MHSI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power; and

16-93

b. A small break LOCA event, with a loss of offsite power.

~~In the event of a large break LOCA, when the only available LHSI connection is located adjacent to the broken cold leg, ECCS delivery to the reactor vessel downcomer may be affected by steam entrainment to the broken leg. This assumes that one train is out of service due to preventative maintenance, one train is assumed to have a single failure, and another train feeds the broken loop. In order to mitigate the effect of degraded ECCS delivery due to steam entrainment, isolable ECCS cross-connects are provided. This arrangement directs a portion of the LHSI flow to an adjacent train, thereby reducing flow lost to steam entrainment. The ECCS cross-connects between Trains 1 and 2 and Trains 3 and 4 are normally isolated by two motor-operated valves in series to maintain train separation. Both cross-connect isolation valves are opened when an ECCS train is taken out of service for maintenance and power removed from the motor operators.~~ ECCS cross-connects are provided for the LHSI trains to allow the removal of one train from service. Cross-connects are normally isolated by two motor-operated valves (MOVs) arranged in a series configuration. The MOVs provide separation of trains when the valves are closed. The cross-connects provide the ability to cross-tie between Trains 1 and 2 or between Trains 3 and 4. The cross-connects do not provide the ability to cross-tie between any other train combinations. When the cross-connects are opened and the

BASES

APPLICABLE SAFETY ANALYSES (continued)

associated trains are cross-tied, the electrical power is required to be removed from the MOVs to ensure the valves remain in the required position.

If Train 1 is removed from service, the MOVs between Train 1 and 2 are opened, and additionally the MOVs between Trains 3 and 4 are also opened. With this configuration, if a DBA were to occur Train 2 would provide 50 percent of the required design flow to RCS loop 1, and 50 percent to RCS loop 2. For train 3 and Train 4, 100 percent of the required design flow would be provided to RCS loops 3 and 4. If Train 3 was lost because of a single failure, Train 4 would split its flow and provide 50 percent to RCS loop 3 and 50 percent to RCS loop 4. In this configuration, no matter in which RCS loop the LOCA occurred a minimum of 150 percent of the required design flow would be delivered to the reactor core by the remaining two trains.

16-93

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boil off rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the MHSI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the MHSI pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling. The cooling capability of the steam generators is enhanced by the operation of the secondary side main steam relief trains.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

~~Four~~ In Modes 1, 2 and 3, three 100% capacity independent (cross-connect closed) ECCS trains are required to ensure that sufficient ECCS flow is available. One of the four ECCS trains is assumed unavailable due to maintenance. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

16-93

BASES

LCO (continued)

An ECCS train consists of an MHSI subsystem, and an LHSI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of injecting upon an PS signal.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the ECCS pumps to the individual cold leg injection nozzles. In the long term, this flow path may be switched to supply its flow to the RCS hot and cold legs.

16-93

The IRWST is sized to ensure~~s~~ that an adequate ~~supply~~volume of borated water is available ~~to cool and depressurize the containment~~ in the event of a ~~Design Basis Accident (~~DBA~~), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS pump operation~~. The IRWST is addressed in LCO 3.5.4.

~~To be considered OPERABLE, the IRWST must meet the water volume and boron concentration limits established in the SRs.~~

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The MHSI pump performance requirements are based on a small break LOCA. MODE 2, and 3 requirements are bounded by the MODE 1 analysis.

16-85

This LCO is only applicable in MODE 3 and above. Below 356°F, the PS signal setpoint for the LHSI subsystem is manually bypassed by operator control (in order to allow the alignment of the LHSI subsystem in RHR mode), and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "LHSI / RHR and Coolant Circulation - High Water Level," and LCO 3.9.5, " LHSI / RHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one MHSI train inoperable, the inoperable components must be returned to OPERABLE status within 120 days. The 120 day Completion Time is based on the assumption in the FSAR Chapter 15 analysis that one ECCS train is assumed out of service for maintenance at the time of the accident.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

B.1 and B.2

With one LHSI train inoperable, an acceptable ECCS configuration can be achieved by opening both ECCS cross connections. In the event of a cold leg break, one train is assumed lost due to steam entrainment to the broken loop, one train is assumed to mitigate the event, ~~one train is assumed to spill out the break,~~ and one train is assumed to have a single failure. A reliability analysis (Ref. 5) has shown that the impact of having only one full ECCS train ~~in~~operable is sufficiently small to justify continued operation for 72 hours. The 120 day Completion Time is based on the assumption in the FSAR Chapter 15 analysis that one ECCS train is assumed out of service ~~for maintenance at the time of the accident~~during the DBA.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6

Periodic inspections of the suction inlet from the IRWST ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR Section 6.2.~~, "Containment Systems."~~
4. FSAR Chapter 15.~~, "Accident Analysis."~~
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

16-73.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the required volumes for an outage and is therefore not limiting. The minimum IRWST volume is determined by ECCS pump NPSH requirements. The minimum boron concentration and isotopic inventory are explicit assumptions in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The importance of its value is small due to the Extra Borati~~on~~ng System (EBS) with its high boron concentration.

16-100

The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the IRWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

For a large break LOCA analysis, the minimum water volume of 500,342 gallons and the lower boron concentration limit of 1700 ppm of ≥~~>~~ 37% enriched boron are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core. This minimum volume bounds the ECCS pump NPSH requirements.

The maximum water volume of 523,703 gallons and the upper limit on boron concentration of 1900 ppm are used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

The upper temperature limit of 122°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on IRWST water temperature are used to maximize the total energy release to containment.

The minimum temperature ~~valve~~ value of 59°F is consistent with mechanical requirements, particularly reactor pressure vessel brittle fracture risk.

The IRWST satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

LCO

The IRWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS pump operation.

16-94

To be considered OPERABLE, the IRWST must meet the temperature, water volume, and boron concentration and enrichment limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, IRWST OPERABILITY requirements are dictated by ECCS OPERABILITY requirements. Since the ECCS must be OPERABLE in MODES 1, 2, 3, and 4, the IRWST must also be OPERABLE to support its operation. In MODES 5 and 6, the IRWST is in standby. 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, "LHSI/RHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "LHSI/RHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

16-94 & 16-95

With IRWST temperature, water volume, boron concentration, or enrichment not within limits, it must be returned to within limits within 8 hours. Under these conditions the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit ~~limit to restore the IRWST boron concentration or enrichment to within limits was developed considering the time required to change the boron concentration/isotopic inventory and the fact that the contents of the tank are still available for injection~~ is acceptable considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

B.1

With the IRWST inoperable for reasons other than Condition A ~~(e.g., water volume)~~, it must be restored to OPERABLE status within 1 hour.

In this Condition, the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the IRWST is not required. The short time limit of 1 hour to restore the IRWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

16-100

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Extra ~~Boration~~ Borating System (EBS)

BASES

BACKGROUND

The EBS is a manually actuated, safety-related system that is used in the mitigation of design basis accidents, including a steam generator tube rupture (SGTR). During this event, the EBS injects boron into the Reactor Coolant System (RCS) to maintain the core subcritical while the RCS is being cooled to the point where the Low ~~Heat~~ Head Safety Injection System can be connected to remove core decay heat. The EBS also provides RCS makeup to balance a portion of the shrinkage during cooldown. The EBS can be used for hydrostatic testing of the RCS but otherwise does not perform any function supporting normal plant operation.

The EBS consists of two identical trains. Each train is composed of its own boron tank, a high pressure 100% capacity pump, a test line, and injection lines to the RCS. The volume of concentrated boric acid required to maintain subcriticality is divided between the two EBS tanks. A common suction header allows either EBS pump to take suction from both tanks. The boron tanks and the primary train lines are filled with borated water and are located in a temperature controlled room to prevent crystallization of the boron (Ref. 1 ~~and 2~~). Outside of the temperature controlled rooms, the EBS piping is filled with lower concentration borated water from the In-Containment Refueling Water Storage Tank.

16-73.1

APPLICABLE SAFETY ANALYSES

If needed, the EBS is manually initiated. A 30 minute operator action time is assumed in the analysis. Once started for safety reasons, the EBS will remain in operation until the boron concentration needed for cold shutdown is reached.

The EBS is initiated for an SGTR to ensure adequate boration to prevent criticality. The contents of the EBS are not credited for core cooling or immediate boration in the LOCA analysis. The EBS maximum boron concentration of 7300 ppm is used in the Boron Precipitation Assessment (Ref. 2). The minimum boron concentration of 7000 ppm is credited in the SGTR analysis and for cooldown from other design basis events. Boron used in the EBS is enriched to ≥ 37% in the B-10 isotope.

The EBS minimum water volume limit of 2345 ft^3 total between the two EBS tanks is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to maintain the core in a shutdown condition following an SGTR or during cooldown for other Design Basis Accidents (DBAs). This volume includes approximately 175 ft^3 of unusable volume in each tank.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum temperature limit of 68°F for the EBS borated water assures that the solution does not reach the point of boron crystallization.

The EBS satisfies Criteri~~a~~on 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO establishes the minimum requirements as well as requirements for contained volume, boron concentration, boron enrichment, and temperature of the EBS inventory (Ref. 3). This ensures that an adequate supply of borated water is available in the event of an SGTR or other design basis event to maintain the reactor subcritical following these accidents.

To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, boron isotopic inventory, and temperature must be met.

APPLICABILITY

In MODES 1, 2, 3, and 4, the EBS is needed to maintain the core subcritical following an SGTR and during cooldown to MODE 5 for DBAs.

An SGTR and other DBAs that rely on the EBS for cooldown are not postulated in MODES 5, and 6 and EBS OPERABILITY is not required.

ACTIONS

A.1

If the boron concentration or boron enrichment of one or both EBS tanks is not within limits, it must be returned to within limits within 72 hours. Because of the low probability of an SGTR or other DBAs, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration or enrichment to within limits.

16-97

Boron concentration or boron enrichment can be restored within the 72 hour limit by mixing fresh boric acid in the Reactor Boron Water Makeup System (RBWMS) and transferring it to the EBS tanks.

B.1

If one EBS train is inoperable for reasons other than Condition A, the inoperable train must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE train is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE train could result in reduced EBS shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE train capable of performing the intended EBS function and the low probability of a DBA occurring.

BASES

ACTIONS (continued)

Certain failure modes of EBS could affect one or both trains of EBS since 100% of the volume of concentrated boric acid required to maintain subcriticality is divided between the two EBS tanks. Both tanks are needed to maintain OPERABILITY. Since the two EBS tanks are interconnected through a normally open flow path that allows either EBS pump to draw from both tanks, failure modes of EBS must consider if both trains are affected. Failure of active mechanical components such as the pump or motor operated valves would only affect one train and can be restored within the 7 day limit.

16-97

C.1

If both EBS trains are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA occurring.

Certain failure modes of EBS could affect one or both trains of EBS. Since 100% of the volume of concentrated boric acid required to maintain subcriticality is divided between the two EBS tanks is needed to maintain OPERABILITY. Since the two EBS tanks are interconnected through a normally open flow path that allows either EBS pump to draw from both tanks, failure modes of EBS must consider if both trains are affected.

A decrease in volume would affect both trains. The likely causes of the EBS tank volume decreasing below limits are leakage, sampling, or testing. The margin for EBS tank volume would make it unlikely that such a decrease in volume would occur without detection. Fresh concentrated boric acid can be supplied from the Reactor Boron Water Makeup System (RBWMS) and volume can be restored within the 8 hour limit.

16-97

The likely cause of the EBS tank temperature decreasing below limits is abnormally low outside ambient temperature coupled with a failure in the safety-related portion of the Fuel Building Ventilation System to maintain EBS room temperatures. The margin for EBS tank temperature to decrease before boron precipitation begins allows a reasonable time to either restore the Fuel Building Ventilation System or start the EBS pump to allow recirculation to heat the EBS tank and temperature can be restored within the 8 hour limit.

D.1 and D.2

If any Required Action and associated Completion Time is not met, the

16-98

BASES

ACTIONS (continued)

unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within ~~12~~ 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach MODE 5 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 24 hours that each EBS tank water temperature and EBS pump room is at or above the specified minimum temperature is frequent enough to identify a temperature change that would approach the acceptable limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. This Frequency has been shown to be acceptable through operating experience.

16-99

The EBS is designed to prevent boric acid crystallization during standby without heat tracing. The temperature inside the Fuel Building where the tanks and main lines are located is maintained above 68°F by safety-related portion of the Fuel Building Ventilation System.

SR 3.5.5.2

Verification every 7 days that the EBS contained volume is above the required limit is frequent enough to assure that this volume will be available for quick injection into the RCS. If the volume is too low, the EBS would not provide enough borated water to ensure subcriticality during recirculation. Since the EBS volume is normally stable, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.5.7

This Surveillance ensures that there is a functioning flow path from the EBS tank to the RCS. An acceptable method is to test the flow path in several separate tests. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance test when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

16-96

SR 3.5.5.8

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 an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. The 24 month Frequency is based on engineering judgment and is considered adequate given that the isolation is not challenged during a SGTR and power operated outboard containment isolation valve can be de-energized and opened manually. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR ~~Chapter 6~~Section 6.8.
2. FSAR ~~Chapter 15~~Section 15.6.3.
3. 10 CFR 50.46.

16-73.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

16-116

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the full flow 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 are pneumatically operated spring closed valves that will fail on the loss of air. The inborad and outboard isolation valves are powered from separate electrical trains and connected to separate control trains.

The full flow purge valves are designed to close in the environment following a LOCA or MSLB. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

The low flow purge valves may be opened during normal operation. In this case, the single failure criterion remains applicable to the low flow purge valves due to failure in the control circuit associated with each valve. The 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 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 valves' 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 39 inch full flow purge valves must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in FSAR Section 6.2.4 (Ref. 2).

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/devices are those listed in Reference 2.

BASES

ACTIONS (continued)

16-116

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 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. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. 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 or more containment isolation valves in one or more penetration flow paths inoperable, except for purge valve leakage not within limit, 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 control and the probability of their misalignment is low.

BASES

ACTIONS (continued)

to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. 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 ~~full flow~~containment purge valves in one or more penetration flow paths are not within the ~~full flow~~containment purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path 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, closed manual valve, or blind flange. A ~~full flow~~ 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 ~~full flow~~containment purge valve remains closed so that a gross breach of containment does not exist.

16-116

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 that those isolation devices outside containment capable of being mispositioned are in the correct position. 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.

16-116

For the containment~~full flow~~ purge valve with 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~~full flow~~ 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

BASES

ACTIONS (continued)

initiative, Generic Issue B-20 (Ref. 4). 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.

Required Action D.2 is modified by two Notes. Note 1 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. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

E.1 and E.2

If any Required Action and associated Completion Time is 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

16-116

Each 39 inch full flow 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 full flow purge valve. The full flow purge valves are designed to close in the environment following a LOCA. However, the DBA dose analysis assumes that each full flow purge line is isolated 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 or by removing the air supply to the valve operator. 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. 5), related to containment purge valve use during plant operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.

BASES

REFERENCES:

1. FSAR ~~Chapter~~ Section 15.0.
2. FSAR Section 6.2.4.
3. NUREG 0800, Section 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
5. Generic Issue B-24, "Seismic Qualification of Electrical and Mechanical Components."

16-73.1

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 transients which result in a negative pressure.

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 and may result in leakage greater than assumed in the accident analysis.

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, relative to containment pressure, are the LOCA and MSLB, which are analyzed using computer pressure transients. The worst case MSLB generates larger mass and energy release than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

16-58

The initial pressure condition used in the containment analysis was 15.96 psia (1.26 psig). This resulted in a maximum peak pressure from a LOCA of 52.0 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure~~,~~ results from the limiting ~~LOCA~~MSLB. P_a is conservatively set at 55 psig. The maximum containment pressure resulting from the worst case LOCA, 52.0 psig, does not exceed the containment design pressure, 62 psig.

The containment was also designed for an external pressure load equivalent to -3.0 psig. An inadvertent actuation of the Severe Accident Heat Removal System is not considered a credible event for the U.S. EPR since it is manually actuated for beyond design basis events only. An evaluation of a Containment cooldown event determined a worse case pressure of -2.9 psig.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on 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

1. FSAR Section 6.2.

16-59

~~2. 10 CFR 50, Appendix K.~~

BASES

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

16-115

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, ~~a weighted~~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 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.

REFERENCES

1. FSAR Section 6.2.
2. FSAR Section 3.8.
3. 10 CFR 50.49.
4. FSAR Section 3.11.
5. FSAR Section 15.0.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Shield Building

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the Containment Building. Between the Containment Building and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

16-113

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the shield building and the containment building. Filters in the system then control the release of radioactive contaminants to the environment. The description of the AVS is provided in the Bases for Specification 3.6.7. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS in the event of a Design Basis Accident.

To ensure the retention of containment leakage within the Containment Building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

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

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The shield building satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

The LCO is modified by a Note allowing the shield building boundary to be opened intermittently under administrative controls. This Note only applies to openings in the shield building boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative

BASES

SURVEILLANCE REQUIREMENTS (continued)

pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.6.2

Maintaining shield building OPERABILITY requires verifying each access opening door is closed. However, all shield building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.6.3 and 3.6.6.4

The Annulus Ventilation System (AVS) exhausts the annulus atmosphere to the environment through appropriate treatment equipment. Each safety AVS train is designed to draw down the annulus to a negative pressure of ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the annulus at a negative pressure ≥ 0.25 inches wg. To ensure that all fission products released to the annulus are treated, SR 3.6.6.3 and SR 3.6.6.4 verify that a pressure in the annulus that is less than the lowest postulated pressure external to the shield building boundary can be established and maintained. When the AVS System is operating as designed, the establishment and maintenance of annulus pressure cannot be accomplished if the shield building boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.6.3, which demonstrates that the annulus can be drawn down to a negative pressure ≥ 0.25 inches wg using one AVS train. SR 3.6.6.4 demonstrates that the annulus can be maintained at a negative pressure ≥ 0.25 inches wg using one AVS train at a flow rate ≤ ~~1320~~1295 cfm. The primary purpose of these SRs is to ensure annulus boundary integrity. The secondary purpose of these SRs is to ensure that the AVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the AVS System. These SRs need not be performed with each safety AVS train. The AVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.6.7, either safety AVS train will perform this test. The inoperability of the AVS System does not necessarily constitute a failure of these Surveillances relative to the shield building OPERABILITY. Operating experience has shown the shield building boundary usually passes these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

16-114

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-124

SR 3.6.6.5

This SR would give advance indication of gross deterioration of the concrete structural integrity of the Shield Building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown.

REFERENCES None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Annulus Ventilation System (AVS)

BASES

BACKGROUND

The AVS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the Containment Building into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The Containment Building is surrounded by a secondary containment called the shield building, which is a concrete structure. Between the Containment Building and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

The AVS maintains a negative pressure in the annulus between the shield building and the Containment Building during operation. Filters in the system control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the AVS. The AVS is designed to permit appropriate periodic pressure and functional testing to assure component integrity, OPERABILITY of active components, and operational performance of the system as required by GDC-43 "Testing of Containment Atmosphere Cleanup Systems" (Ref ~~4~~2).

16-112

The AVS consists of one normal operation filtration train (non-safety related), and two independent and redundant accident filtration trains (safety related). The normal filtration train operates during normal plant operation, including cold shutdown and outages. The normal operations filtration train maintains a negative pressure ≥ 0.25 inches water gauge in the annulus during normal operation. During normal plant operation, the accident filtration trains are not required to be in operation, however they are both available for back-up if the normal filtration train is not able to maintain sufficient negative pressure in the annulus.

During normal operation, the conditioned air is drawn from the Nuclear Auxiliary Building Ventilation supply shaft to the bottom of annulus through a fire damper, manual regulated control damper, and two motor operated isolation dampers. The exhaust air is drawn through a vent at the top of annulus through two motor operated isolation dampers and fire dampers to the Nuclear Auxiliary Building Ventilation system exhaust fans via air shaft cell 3. See FSAR Section 9.4.3 (Ref. ~~5~~3). The exhaust air from cell 3 is filtered by the pre-filter and HEPA filter and then discharged through the vent stack. The annulus air inlet and exhaust motor operated

BASES

BACKGROUND (continued)

16-112

isolation dampers of the normal filtration train are the only components which are safety related. The four safety-related class motor operated air tight dampers will isolate the annulus from the non-safety normal operation train in case of a design basis accident. The two isolation dampers in both the supply and exhaust train are powered by separate divisions and are supplied by the emergency diesel generators. Each isolation damper can be operated either automatically or manually from the Main Control Room.

In normal operation mode, if there is a loss of negative pressure in the annulus, failiure of the Nuclear Auxiliary Ventilation System, or Loss of Offsite Power, the normal operation filtration train is considered lost and one of the accident filtration trains is switched on. The two isolation dampers on both the normal supply and exhaust trains are closed and one of the two accident filtration trains is switched on. The other accident filtration train is available for backup.

16-112

The AVS accident filtration trains are used during a design basis event to contain leaks from the primary containment by maintaining a negative pressure in the annulus. During a design basis event, the annulus air is filtered before releasing to the environment. There are two independent 100% accident trains. Each train consists of an upstream air-tight motor controlled damper, electrical heater, pre-filter, upstream HEPA filter, an activated charcoal adsorber for removal of radio-iodines, downstream HEPA filter, downstream air-tight motor controlled damper, fan, and back-draft damper. The upstream HEPA filter removes the fine discrete particulate matter from the air stream. The downstream HEPA filter following the charcoal adsorber ~~The downstream bank of HEPA filters following the charcoal adsorber~~ collects carbon particles and provides backup in case of failure of the upstream~~main~~ HEPA filter ~~bank~~. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the ~~s~~Shield ~~b~~Building by means of filtered exhaust ventilation of the ~~s~~Shield ~~b~~Building following receipt of a containment isolation signal. The system is described in Reference ~~2~~4.

16-60

The prefilters remove large particles in the air~~, and the moisture separators remove entrained water droplets present,~~ to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters ~~may be included to~~ reduce the relative humidity of the airstream to 70 percent or less~~on systems that operate in high humidity~~. Monthly~~Continuous~~ operation of each train, for ≥ 15 minutes~~at least 10 hours per month~~, with heaters on, reduces moisture buildup on the~~ir~~ HEPA filters and adsorbers. The heater operation time and monthly Frequency are consistent with Reference 6.

BASES

BACKGROUND (continued)

16-112

During normal operation, the AVS normal operation filtration train (non-safety related) maintains a negative pressure in the annulus and processes the air through HEPA filters.

The isolation dampers on the normal operation filtration train and the accident filtration trains can be operated either automatically or manually from the Main Control Room.

The AVS accident filtration train reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the AVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The AVS design basis is to mitigate the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. ~~3~~5) assumes that only one train of the AVS is OPERABLE due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. For all events analyzed, the AVS is assumed to be automatically initiated to reduce via filtration and adsorption, the radioactive material released to the environment.

The modeled AVS actuation in the safety analyses is based upon a worst case response time following a containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of 0.25 inches wg in the shield building, is 305 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The AVS satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

In the event of a DBA, one AVS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two safety related trains of the AVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor

BASES

APPLICABILITY (continued)

shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the AVS is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

16-112

With one AVS accident filtration train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1

If one or more normal operation train motor operated dampers is inoperable, the annulus ventilation negative pressure could not be established or maintained in the event of an accident. Therefore damper OPERABILITY must be restored in 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of the containment and the low probability of a Design Basis Accident occuring during this time period.

~~B~~C.1 and ~~B~~C.2

If the AVS accident filtration train or isolation damper 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.

16-112

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

~~Operating each AVS train for ≥ 10 continuous hours ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the two train redundancy available.~~

SR 3.6.7.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). 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.

16-112

SR 3.6.7.3

The automatic startup ensures that each AVS accident filtration train responds properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the

BASES

SURVEILLANCE REQUIREMENTS (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.7.1.

16-112

SR 3.6.7.4

This SR verifies that the normal operation train motor operated isolation dampers close on an actual or simulated isolation signal. This SR demonstrates that a closed annulus volume can be established in the event of a Design Basis Accident when a negative pressure in the annulus is credited in the dose analysis. The 24 month Frequency is based on the need to perform this Surveillance in conjunction with SR 3.6.7.3.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. 10 CFR 50, Appendix A, GDC 43.

3. FSAR Section 9.4.3.

~~2~~4. FSAR Section 6.2.3.

~~3~~5. FSAR Chapter 15.

16-60

6. Regulatory Guide 1.52, Rev. 3.

~~4. 10 CFR 50, Appendix A, GDC 43.~~

~~5. FSAR Section 9.4.3.~~

BASES

ACTIONS

A.1 and A.2

16-86

With one required MSSV inoperable, the associated MSRT is verified OPERABLE and action must be taken to restore the valve to OPERABLE status within ~~30 days~~72 hours. Verification of MSRT OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSRT is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSRT. If the OPERABILITY of the associated MSRT cannot be verified, however, Condition ~~C~~B must be immediately entered.

~~The 30 day Completion Time considers the following:~~

~~a. With one MSSV inoperable, the resulting relief capacity of the affected SG is 75~~ Copy Deleted ~~nt the MSRT) of the full load steam generation~~ ~~m generator, which is greater than the 50% reli~~ ~~d in the safety analysis.~~

~~b. The remaining OPERABLE overpressure protection devices are sufficient for heat removal for the long term phase.~~

The ~~30 day~~72 hour Completion Time is considered reasonable ~~for restoring the inoperable components to OPERABLE status~~since the remaining relief capacity is adequate to mitigate all postulated events.

~~B.1~~

~~With two MSSVs inoperable, actions must be taken to restore the inoperable MSSVs to OPERABLE status within 7 days.~~

~~This Completion Time~~ Copy Deleted ~~se:~~

~~a. With two MSSVs inoperable on the same SG, the resulting relief capacity of the affected SG is 50% (taking into account the MSRT) of the full load steam generation per SG, which is equal to 50% relief capacity considered in the safety analysis.~~

~~b. With one MSSV inoperable on one SG and one MSSV inoperable on another SG, the resulting relief capacity for each SG is 75% of the full load steam generation per SG. This combination of inoperabilities is different from the one in the safety analysis. However, the total relief capacity of the four SGs is 350% of the full load steam generation per SG, which is the exact relief capacity considered in the safety analysis.~~

16-86 Copy Deleted

~~c. The remaining OPERABLE overpressure protection devices (MSRTs) are sufficient for heat removal in the long term phase.~~

BASES

ACTIONS (continued)

~~The 7 day Completion T~~ Copy Deleted ~~asonable based on operating experience to accompli~~ ~~n in an orderly manner without challenging unit~~

~~C~~B.1 and ~~C~~B.2

16-86

If any Required Action and associated Completion Time cannot be met or if ~~three~~ two or more MSSVs 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 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 setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5).

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. The SR allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. 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.

BASES

REFERENCES

1. FSAR Section 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM-1-1987.

16-91

~~6. FSAR Section 15.4.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

16-108

The limiting case for the containment analysis is the MSLB inside containment, with offsite power available, and failure of the MSIV on the affected steam generator to close. At lower power levels, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control 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 boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different MSLB events against different acceptance criteria. The double-ended guillotine break of a main steam line in the valve compartment in the Safeguards Building 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 MSLB outside containment, upstream of an MSIV at hot zero power is the limiting case for a post trip return to power. The analysis includes a spectrum of break sizes, 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 is delayed. The worse case single failure is a main steam relief control valve associated with one of the unaffected steam generators failed in the fully open position (Ref. 3). 16-108

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

16-108

b. A break outside of containment and upstream from the MSIVs 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 will be isolated by the closure of the MSIVs.

~~c~~d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

~~d~~e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, two control lines per MSIV are OPERABLE, and they close on an isolation actuation signal.

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 50.34 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed and the steam generator energy is low.

In MODE 5 or 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.3

This SR verifies that MSIV closure time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs can not be full stroke tested when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. 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, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.4

16-118

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 24 months on a STAGGERED TEST BASIS for each control line. As a result, each MSIV will have a closure test on a 24 month Frequency. The test will alternate use of the two redundant control lines. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint. This SR is modified by a NOTE that requires the performance of this surveillance ~~prior to entry into MODE 2~~only in MODES 1 and 2.

BASES

REFERENCES

1. FSAR Section 10.3.
2. FSAR Section 6.2.
3. FSAR ~~Chapter 15~~Section 15.1.5.
4. 10 CFR 50.34.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

16-73.1

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater (MFW) Valves

BASES

BACKGROUND

On each of the four steam generators (SGs), the Main Feedwater valves (MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), FW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs)) are located in valve stations, physically separated from each other and from other systems. Within these valve compartments, the MFW lines are arranged in three trains, one Very Low Load, one Low Load, and one Full Load train. The full load flow path for each steam generator includes one MFWFLCV, one MFWFLIV, and the MFWMIV. The low load flow path for each steam generator includes one MWFLLCV, one MFWLLIV, and the MFWMIV. The very low load flow path for each steam generator includes one MFWVLLCV, one MFWLLIV, and the MFWMIV. Each of these trains can be isolated redundantly by one isolation valve, one control valve, or the MFWMIV. The Low Load isolation valve allows isolation of the Low Load and the Very Low Load train at the same time.

The closure of these valves allows limiting the filling of the steam generators in case of a too high feedwater flowrate which could impair the functioning of the safety valves of the Main Steam System.

In the event of a secondary side pipe rupture inside containment, the valves also limit the quantity of high energy fluid that enters containment through the break and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact loops. They also reduce the cooldown effects in case of Main Steam Line Breaks (MSLBs) or in case of excessive increase in feedwater flowrate caused by a feedwater system malfunction.

16-92

A MFW Isolation valve outside containment and a MFW check valve inside containment provide isolation of the containment ~~isolation function~~.

The MFWFLIVs and MFWFLCVs close on a reactor trip. The MFWMIVs and low and low-low range control and isolation valves close in response to steam generator level as described in Reference 1. ~~The MFWMIV closes on a containment isolation signal.~~ The MFW valves may also be actuated manually.

A description of the MFW valves is found in FSAR Section 10.4.7 (Ref. 1).

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low and all MFW valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MFW flow path.

A.1

With one valve in the full load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 7 day Completion Time is reasonable, based on operating experience.

16-92

B.1 and B.2

With two valves in the full load flow path inoperable, close or isolate the inoperable affected flow path in 72 hours. When the flow path is isolated, it is performing its required safety function.~~action must be taken to restore the affected valves to OPERABLE status within 72 hours.~~
The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves~~associated MFWFLCV, MFWFLIV, and MFWMIV~~; and the low probability of an event that would isolation of the the main feedwater flow paths occurring during this ~~time~~ period.~~, that would require isolation of the MFW full load flow path.~~
For inoperable full load flow path valves that cannot be restored to OPERABLE status within the specified~~The 72 hour~~ Completion Time but are closed or isolated, the flow paths must be verified on a periodic basis to be closed or isolated. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is reasonable, based on ~~operating experience~~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

With three valves in the full load flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, one valve in the affected flow path must be restored to OPERABLE status, or the affected flow path

BASES

ACTIONS (continued)

16-92

isolated within 8 hours. This action return the system to the situation in which at least one valve in the affected flow path is performing the required safety function. The 8 hour Completion Time is a reasonable amount of time to complete the actions required to isolate the main feedwater flow path which includes performing a controlled plant shutdown. The Completion Time is reasonable based on operating experience to reach MODE 2 with the main feedwater flow path closed, from full-power conditions in an orderly manner and without challenging plant systems.~~action must be taken to restore the affected valves to OPERABLE status within 8 hours. The 8 hour Completion Time takes into account the redundancy afforded by the redundant actuation trains on MFW full load flow path valves and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 8 hour Completion Time is reasonable based on operating experience.~~

D.1 and D.2

With one or more MFWLLIVs, MFWLLCVs, or MFWVLLCVs in the low load or very low load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 8 hours or isolate the flow path. When the valves are closed, they are performing their required safety function

Inoperable MFW low load and very low load flow path valves that are closed as a result of this Required Action, must be verified on a periodic basis that they are 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, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

E.1 and E.2

If the MFW valves cannot be restored to OPERABLE status or are not 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 at least in 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 MODE 2 conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each valve can close on an actual or simulated actuation signal. This Surveillance is normally performed during shutdown or upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 10.4.7.
2. ASME $\boxed{2004}$ Code for Operation and Maintenance of Nuclear Power Plants.

16-92

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water (ESW) System

BASES

BACKGROUND

The ESW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the ESW System also provides this function for the associated safety related and nonsafety related systems. The safety related function is covered by this LCO.

The ESW System consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, pump, piping, valving, instrumentation, and mechanical filtration. Each safety related 2-cell seismic Category I mechanical draft cooling tower rejects energy from the ESW fluid to the ambient and returns the cooled fluid to the ESW cooling tower basin, from which the ESW pumps take suction. Each ESW cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA. The system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

The mechanical draft cooling towers and basins are safety related, seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

[The seismic Category 1 makeup necessary to support 30 days of post accident mitigation is site specific and details are to be provided by the Combined License applicant].

16-89

Additional information about the design and operation of the ESW System along with a list of the components served, is presented in FSAR Section 9.2.1 (Ref. 1) and FSAR Section 9.2.5 (Ref. 5). The principal safety related functions of the ESW System is the removal of decay heat from the reactor and reactor coolant pump thermal barrier cooling via the Component Cooling Water (CCW) System and removal of operational heat from the emergency diesel generator (EDG).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ESW System is for two ESW trains, in conjunction with the CCW System, to remove core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System pumps. The ESW System also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident.

The ESW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in FSAR 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 CCW and RHR loops that are operating. Two ESW trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESW System temperature of 95°F occurring simultaneously with maximum heat loads on the system.

Each ESW basin is sized for 3 days of post LOCA operation without requiring makeup. ESW basin makeup is required to maintain NPSH for the ESW pumps beyond 3 days. This volume of water is assumed to be at ≤ 90°F during normal plant operation to prevent exceeding the maximum ESW temperature during a LOCA.

16-89

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference ~~1~~5 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The ESW cooling tower and basin is designed in accordance with Regulatory Guide 1.27 (Ref. 4), which requires a 30 day supply of cooling water in the ESW basin, or equivalent make-up.

The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The ESW System consists of four trains. Four ESW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An ESW train is considered OPERABLE when two cooling tower fans, pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the ESW basin contains ≥ 27.2 feet of water at ≤ 90°F with capability from makeup from OPERABLE source. [COL applicant to provide definition of OPERABLE makeup source.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.6

This SR verifies proper automatic operation of the ESW pumps and cooling tower fans on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 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. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.7

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The 24 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. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. This SR verifies that the ESW makeup flowrate is ≥ 300 gpm.

REFERENCES

1. FSAR Section 9.2.1.
2. FSAR Section 6.2.
3. FSAR Section 5.4.7.
4. Regulatory Guide 1.27.
5. FSAR Section 9.2.5.

16-89

BASES

LCO (continued)

stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to ensure that the CRE will remain habitable during and following a postulated accident (i.e., LOCA, main steam line break, rod ejection, and fuel handling accident).

16-63

In MODE 5 or 6, the CREF is also required to cope with a failure of the Gaseous Waste Processing System.

During movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

With one CREF train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the OPERABLE CREF train is adequate to perform the CRE occupant protection function. However, the overall system reliability is reduced. The 7 day Completion Time is based on the low probability of a postulated accident occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of postulated accident consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body 5 rem TEDE), or inadequate protection of CRE occupants from ~~[[~~hazardous chemicals or~~]]~~ smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within ~~90~~ 60 days

16-63

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological ~~[[~~or chemical~~]]~~ event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a postulated accident, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of postulated accident

BASES

ACTIONS (continued)

16-63

consequences, and that CRE occupants are protected from ~~[[~~hazardous chemicals and ~~]]~~smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of mitigating actions. The ~~90~~ 60 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a postulated accident. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if any Required Action and Completion Time of Condition A or B cannot be met, the unit must be placed in a MODE that minimizes 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.

D.1 and D.2 16-63

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREF train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREF train in the emergency mode. This action ensures that the other train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

Required Action D.1 is modified by a Note indicating to place the system in the toxic gas isolation state with outside air isolated.

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains inoperable, or with one or more CREF trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

16-63

F.1

With both Iodine Filtration trains and associated Air Conditioning trains inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREF 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 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy. The heater energization time and 31 day Frequency are consistent with Reference 8.

16-60

SR 3.7.10.2

This SR verifies that the required CREF train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, carbon adsorber efficiency, minimum flow rate, and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

REFERENCES

1. FSAR Section 9.4.

2. ~~Chapter 15~~FSAR Section 15.6.

16-73.1

3. FSAR Section 6.4.

4. FSAR Section 9.5.

5. Regulatory Guide 1.196.

6. NEI 99-03, "Control Room Habitability Assessment," March 2003.

7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability" (ADAMS Accession No. ML040300694).

8. Regulatory Guide 1.52, Rev. 3.

16-60

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Verifying that safeguards building and fuel building negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to safeguards building and fuel building pressure variations and pressure instrument drift during the applicable MODES.

SR 3.7.12.2

Maintaining safeguards building and fuel building OPERABILITY requires verifying each access opening door is closed. However, all safeguards building and fuel building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.7.12.3

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

16-60

Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available. The heater energization time and 31 day Frequency are consistent with Reference 7.

SR 3.7.12.4

This SR verifies that the required SBVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, minimum system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.5

This SR verifies that each SBVS train starts and operates on an actual or simulated actuation signal. The 24 month Frequency is consistent with Reference 7.

SR 3.7.12.6 and 3.7.12.7

16-110

The SBVS exhausts the safeguards building and fuel building atmosphere to the environment through appropriate treatment equipment. Each safety SBVS train is designed to draw down the safeguards building and fuel building to a negative pressure ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the safeguards building and fuel building at a negative pressure ≥ 0.25 inches wg at a flow rate ≤ ~~2,400~~2,640 cfm from the safeguards building and fuel building. To ensure that all fission products released to the safeguards building and fuel building are treated, SR 3.7.12.6 and SR 3.7.12.7 verify that a pressure in the safeguards building and fuel building that is less than the lowest postulated pressure external to the safeguards building and fuel building boundaries can be established and maintained. When the SBVS is operating as designed, the establishment and maintenance of safeguards building and fuel building pressure cannot be accomplished if the safeguards building or fuel building boundaries is not intact. Establishment of this pressure is confirmed by SR 3.7.12.6. SR 3.7.12.7 demonstrates that the safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches wg. The primary purpose of these SRs is to ensure safeguards building and fuel building boundary integrity. The secondary purpose of these SRs is to ensure that the SBVS train being tested functions as designed. ~~There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SBVS.~~ These SRs need not be performed with each safety SBVS train. The SBVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.7.12, either safety SBVS train will perform this test. The inoperability of the SBVS does not necessarily constitute a failure of these Surveillances relative to the safeguards building and fuel building OPERABILITY. Operating experience has shown the safeguards building and fuel building boundaries usually pass these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

16-87

19-109

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Boron Concentration and Enrichment

---Reviewer's Note---
The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.

BASES

BACKGROUND

As described in the following LCO 3.7.16, "Spent Fuel Storage," fuel assemblies are stored in the spent fuel racks without restriction. Although the water in the spent fuel pool is normally borated to ≥ [~~1291~~1700] ppm with boric acid enriched to ≥ 37% ~~B^{10}~~B-10, the criteria that limit storage of a fuel assembly to specific locations are conservatively developed without taking credit for boron.

APPLICABLE SAFETY ANALYSES

----------------------------------REVIEWER'S NOTE----------------------------------
The design of the spent fuel storage racks is the responsibility of the COL applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks.

The concentration and enrichment of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The spent fuel pool boron concentration is required to be ≥ [~~1291~~1334] ppm boron enriched to ≥ 37% ~~B^{10}~~B-10. The specified concentration and enrichment of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 1. This concentration of dissolved boron is the minimum required concentration and enrichment for fuel assembly storage and movement within the spent fuel pool.

19-109

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. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration or enrichment of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated 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 fuel storage pool is within the required limit. As long as this SR and SR 3.7.13.2 are met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

SR 3.7.15.2

Verification every 24 months that the ~~B^{10}~~B-10 enrichment is within limit ensures that the ~~B^{10}~~B-10 concentration assumed in the accident analyses is available. Since the boron in the spent fuel pool is not exposed to a significant neutron flux, 24 months is considered conservative.

REFERENCES

1. ANSI/ANS 8.1-1998 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

BASES

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 the secondary specific activity cannot be restored to within limits 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 3.7.17.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic 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.

REFERENCES

1. 10 CFR 50.34.

16-73.1

2. FSAR ~~Chapter 15~~Section 15.1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of once every 72 hours is 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. ~~Chapter 15~~FSAR Section 15.4.

16-73.1

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

16-107

BASES

BACKGROUND	~~The~~ Three source range neutron flux monitors are used during refueling operations and prior to criticality to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the nuclear instrumentation system. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The installed source range neutron flux monitors 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 the lower six decades of neutron flux (1E+6 cps). The detectors also provide visual indication in the control room and can provide an audible count rate to alert operators to a possible dilution accident. The nuclear instrumentation is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	The source range neutron flux monitors have no safety function and are not assumed to function during any design basis accident or transient analysis. However, the source range neutron flux monitors provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in the Technical Specifications. The source range neutron flux monitors satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.
APPLICABILITY	In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels.

BASES

ACTIONS

A.1 and A.2

With one required source range neutron flux monitor inoperable redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, positive reactivity additions including introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration specified in the COLR must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

16-107

B.1~~ and B.2~~

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and 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 time period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.~~3~~2.1 is 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 of 12 hours is based on operating experience that demonstrates channel failure is rare.

SR 3.9.2.2

SR 3.9.~~3~~2.2 is the performance of a CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CALIBRATION. The CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CALIBRATION also includes verification of the audible count rate and alarm function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

~~2. Chapter 15.~~

16-73.1

B 3.9 REFUELING OPERATIONS

B 3.9.3 ~~Containment Penetrations~~Decay Time

16-65

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment or in the fuel handling area inside nthe Auxiliary Building requires allowing at least 34 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that sufficient radioactive decay has occurred in the event of a fuel handing accident (Refs. 1 and 2). Sufficient radioactive decay of short-lived fission products would have occurred to limit offsite doses from the accident to within the values reported in Chapter 15.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the radioactivity decay time is an initial condition design parameter in the analysis of a fuel handling accident inside containment or in the Fuel Building, as postulated by Regulatory Guide 1.183 (Ref. 1).

The fuel handling accident analysis inside containment or in the Fuel Building is described in Reference 2. This analysis assumes a minimum radioactive decay time of 34 hours.

Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum radioactive decay time of 34 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the Fuel Building are within the values calculated in Reference 2.

APPLICABILITY

Radioactive decay time is applicable when moving irradiated fuel assemblies in containment or in the Fuel Building. The LCO minimizes the possibility of radioactive release due to a fuel handling accident that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved, 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 also covered by LCO 3.7.14, “Spent Fuel Storage Pool Water Level” and LCO 3.9.6, “Refueling Cavity Water Level”.

BASES

ACTIONS	A.1 With a decay time of less than 34 hours, all operations involving movement of irradiated fuel assemblies within containment or in the fuel handling area inside the Auxiliary Building shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement to a safe position.
SURVEILLANCE REQUIREMENTS BASES	SR 3.9.3.1 Verification that the reactor has been subcritical for at least 34 hours prior to movement of irradiated fuel in the reactor pressure vessel to the refueling cavity in containment or to the Fuel Building ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Specifying radioactive decay time limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident (Ref. 2).
REFERENCES	1. Regulatory Guide 1.183. 2. FSAR Section 15.0.3.10.

16-65

BASES

ACTIONS

A.1

If the RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Immediate suspension of positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of 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 decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, and A.6.

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured ~~with four bolts~~;

b. One door in each air lock must be closed; and

16-83

BASES

ACTIONS (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Ventilation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The minimum flow rate specified is 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 RHR System.

REFERENCES

1. ~~Chapter 5~~FSAR Section 5.4.7.

16-73.1

BASES

ACTIONS (continued)

B.3, B.4, and B.5

16-83

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured ~~with four bolts~~;

b. One door in each air lock must be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Ventilation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time and the features available to maintain RHR operation and vessel level (Ref. 1).

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The minimum flow rate specified is 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 for monitoring the RHR System in the control room.

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional LHSI 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.

BASES

REFERENCES 1. ~~Chapter 5~~FSAR Section 5.4.7.

16-73.1

BASES

APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are 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, "Spent 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 irradiated fuel assemblies within the containment 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 analysis of the postulated fuel handling accident 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). 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, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.25, March 1972. 2. ~~Chapter 15~~FSAR Section 15.0.3.10. 16-73.1 3. Regulatory Guide 1.183, Table 6, July 2000.