PMComanchePekNPEm Resource

From: Monarque, Stephen
Sent: Tuesday, September 22, 2009 5:13 PM
To: ComanchePeakCOL Resource
Subject: FW: FSAR Rev. 5 and Part 4 Rev. 0 (UTR) nonpublic until SUNSI review completed
Attachments: TXNB-09043 FSAR R5 & Part 4 R0 UTR.pdf

From: nancy.douglas@txu.com [mailto:nancy.douglas@txu.com]
Sent: Wednesday, September 16, 2009 1:05 PM
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Subject: FSAR Rev. 5 and Part 4 Rev. 0 (UTR)

Luminant has submitted the attached FSAR Rev. 5 and Part 4 Rev 0 (UTR). If there are any questions regarding the letter, please contact me or contact Don Woodlan (254-897-6887, Donald.Woodlan@luminant.com).

Thanks,
Nancy Douglas
NuBuild

Hearing Identifier: ComanchePeak_COL_NonPublic
Email Number: 1098

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Subject: FW: FSAR Rev. 5 and Part 4 Rev. 0 (UTR) nonpublic until SUNSI review completed
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Ref. # 10 CFR 52

September 16, 2009

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555
ATTN: David B. Matthews, Director
Division of New Reactor Licensing

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 3 AND 4
DOCKET NUMBERS 52-034 AND 52-035
COMBINED LICENSE APPLICATION UPDATE TRACKING REPORT (FSAR REV. 5 AND COLA PART 4 REV. 0)

Dear Sir:

Luminant Generation Company LLC (Luminant) herein submits the fifth FSAR Update Tracking Report (UTR) for the Comanche Peak Nuclear Power Plant Units 3 and 4 Combined License Application (COLA) and the first UTR for COLA Part 4, Technical Specifications. The marked-up pages reflect responses to Requests for Additional Information resulting from the NRC review of the Mitsubishi US-APWR Design Certification application and from discussions with the NRC in a DCWG meeting on July 16, 2009 (ML092100685).

Should you have any questions regarding these UTRs, please contact Don Woodlan (254-897-6887, Donald.Woodlan@luminant.com) or me.

There are no commitments in this letter.

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

Executed on September 16, 2009.

Sincerely,

Luminant Generation Company LLC

Donald R. Woodlan for

Rafael Flores

Attachment: CD containing COL Application Part 2, FSAR Update Tracking Report Revision 5 and COL Application Part 4, Technical Specifications Update Tracking Report Revision 0

Email Distribution w/ attachment

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September 11 2009

Comanche Peak Nuclear Power Plant, Units 3 & 4
COL Application

Part 2, FSAR

Update Tracking Report

Revision 5

Revision History

Revision	Date	Update Description
0	3/31/2009	Original Issue Updated Chapters: Ch.1, 2, 3, 5, 6, 8, 9, 11, 12, 13, 14, 17 and 19 Incorporated responses to following RAIs: No.1
1	4/24/2009	Updated Chapters: Ch. 2, 6
-	5/1/2009	Updated Chapters: Ch. 1, 5,14 See Luminant Letter no. TXNB-09010 Date 5/1/2009 Incorporated responses to following RAIs: No. 1, 2
2	5/08/2009	Updated Chapters: Ch 1, 2
-	5/26/2009	Updated Chapters: Ch. 7 See Luminant Letter no. TXNB-09020 Date 5/26/2009 Incorporated responses to following RAIs: No. 4, 5
-	6/17/2009	Updated Chapters: Ch. 1,10 See Luminant Letter no. TXNB-09023 Date 6/17/2009 Incorporated responses to following RAIs: No. 6
3	6/30/2009	Updated Chapters: Ch 3 , 9,10,12,14,19
-	8/7/2009	Updated Chapters: Ch. 1, 5, 10

		See Luminant Letter no. TXNB-09028 Date 8/7/2009 Incorporated responses to following RAIs: No. 7, 8
-	8/24/2009	Updated Chapters: Ch. 1, 3, 10 See Luminant Letter no. TXNB-09033 Date 8/24/2009 Incorporated responses to following RAIs: No. 12, 16
-	8/24/2009	Updated Chapters: Ch. 1, 3, 10 See Luminant Letter no. TXNB-09034 Date 8/24/2009 Incorporated responses to following RAIs: No. 17, 20
4	8/28/2009	Updated Chapters: Ch 2, 3, 4, 5, 6, 7, 8, 10, 11, 12, 13, 14
-	8/28/2009	Updated Chapters: Ch. 2 See Luminant Letter no. TXNB-09035 Date 8/28/2009 Incorporated responses to following RAIs: No. 11, 14, 21, 22
5	9/11/2009	Updated Chapters: Ch 9, 11

Chapter 1

Chapter 1 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00586	1.2	1.2-3 1.2-4	Consistent with Subsection 9.4.5.2.6	Add “UHS” before “ESW pump”.	0
CTS-00586	1.2	1.2-4	Erratum	Change the number of pumps.	0
CTS-00534	1.8	1.8-13	Consistent with DCD Rev.1	Correct COL 3.2(4) and 3.2(5) to reflect wording changes in DCD Rev1.	0
CTS-00535	1.8	1.8-16	Consistent with DCD Rev.1	Correct COL3.5(2) to reflect wording changes in DCD Rev1.	0
CTS-00536	1.8	1.8-23	Editorial correction	Change “AD/V2” to “AD/V^2”.	0
CTS-00537	1.8	1.8-28	Consistent with DCD Rev.1	Correct COL3.8(19) to reflect wording changes in DCD Rev1.	0
CTS-00527	1.8	1.8-30	Consistent with DCD Rev.1	Correct COL3.9(2) to reflect wording changes in DCD Rev1.	0
CTS-00538	1.8	1.8-33	Consistent with DCD Rev.1	Correct COL3.10(9) to reflect wording changes in DCD Rev1.	0
CTS-00550	1.8	1.8-41	Editorial correction	Delete “these” from COL 6.2(1).	0
CTS-00539	1.8	1.8-43	Editorial correction	Add “and” in COL 6.4(5).	0
CTS-00540	1.8	1.8-55	Editorial correction	Change “an” to “a ” in COL10.3(1).	0
CTS-00541	1.8	1.8-56	Editorial correction	Change “deta” to “data” in COL11.2(3).	0
CTS-00542	1.8	1.8-61	Consistent with DCD Rev.1	Correct COL12.1(1) to reflect wording changes in DCD Rev1.	0
DCD_12.01-2	1.8	1.8-61	Delete Outdated RG	Delete reference to RG8.20, 8.26, and 8.32 from COL12.1(3).	0
CTS-00543	1.8	1.8-64	Consistent with DCD Rev.1	Correct COL13.1(5), 13.2(2) and 13.2(3) to reflect wording changes in DCD Rev1.	0
CTS-00610	13.5.2	1.8-66	Update	Add Subsection “13.5.2.1” in Table 1.8-201.	0
CTS-00544	1.8	1.8-67	Consistent with DCD Rev.1	Correct COL13.6(1)and 13.7(1) to reflect wording changes in DCD Rev1.	0
CTS-00545	1.8	1.8-70	Consistent with DCD Rev.1	Delete COL16.1_3(1).	0
CTS-00546	1.8	1.8-71	Editorial correction	Delete “and” from COL16.1_3.3.2(1).	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00526	1.8	1.8-74	Consistent with DCD Rev.1	Correct COL17.5(1) to reflect wording changes in DCD Rev1.	0
CTS-00530	1.9	1.9-7	Correct Corresponding Section	Delete reference to 5.2.1.2 from RG1.84.	0
CTS-00529	1.9	1.9-16	Correct COLA/FSAR Status	Add “with exceptions” to “Conformance” in RG 4.15.	0
DCD_12.01-2	1.9	1.9-18 1.9-19	Delete Outdated RG	Delete reference to RG8.20, 8.26, and 8.32 from Table1.9-203.	0
RCOL2_14.03-1	Table 1.8-201	1.8-69	Responses to RAI No. 1 Luminant Letter TXNB-09010 Dated 5/1/2009	Add FSAR location “14.2.12.1.90.C8” as resolution of COL 14.2(10).	-
CTS-00703	Table 1.9-201	1.9-4	To Reflect CPNPP Units 3 and 4 compliance with RG 1.23.	Added “Second Prepared Revision, April 1986” in the Revision/Date category and “revision of record CPNPP Units 1 and 2” to the COLA FSAR Status category.	2
RCOL2_10.02.03-01	Table 1.8-201	1.8-54	Response to RAI No. 6 Luminant Letter no.TXNB-09023 Date 06/17/2009	For COL 10.2(1), replace the word “develop” with “establish a” and delete “and then to implement” in the first sentence. Delete the entire second sentence. Insert “A” under the column “COL Applicant Item”; delete “H” and delete “b” from columns labeled “COL Holder Item” and “Rationale”.	-
RCOL2_10.03.06-2	Table 1.8-201	1.8-55	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Replace the revision number for NSAC-202L from “R3” to “R2”. Insert “and are susceptible to erosion-corrosion damage” at end of 1st sentence for COL 10.3(1).	-
RCOL2_10.03-1	Table 1.8-201	1.8-55	Response to RAI No. 16 Luminant Letter no.TXNB-09033 Date 08/24/2009	Delete COL 10.3(2) description and state “Delete from DCD”.	-

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_01-1	Table 1.7-202	1.7-3	Response to RAI No. 20 Luminant Letter no.TXNB-09034 Date 08/24/2009	Delete Figure 9.2.4-201, “Sanitary Wastewater Treatment System Flow Diagram,” from Table 1.7-202.	-

Chapter 2

Chapter 2 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00636	Table 2.0-1R	2.0-3 2.0-13	Editorial correction	Change “X/Q” to “χ/Q”. (χ is a Greek letter.)	0
CTS-00637	Table 2.2-203 Table 2.2-206	2.2-28 2.2-33	Editorial correction	Change “CPNPP Units 1 & 2” to “CPNPP Units 1 and 2”.	0
CTS-00587	Table 2.3-206	2.3-71	Erratum	Change “5” to “3”.	0
CTS-00636	Table 2.3-342	2.3-252 2.3-253	Editorial correction	Change “X/Q” to “χ/Q”. (χ is a Greek letter.)	0
CTS-00590	2.4.1.1	2.4-2	Editorial correction	Change “grade” to “floor elevation”.	0
CTS-00591	2.4.1.1	2.4-3	Editorial correction	Change “Category I seismic requirement” to “seismic category I requirement”.	0
CTS-00661	2.4.1.2.1	2.4-5	Editorial correction	Add "(Figure 2.4.1-207)" after Morris-Sheppard Dam.	0
CTS-00662	2.4.1.2.1	2.4-6	Editorial correction	Add reference numbers according to CTS-00666.	0
CTS-00592	2.4.1.2.3.2	2.4-7	Editorial correction	Change “intake pumping station” to “makeup water intake structure” and “cooling tower makeup pumps” to “makeup water pumps, makeup water jockey pump”.	0
CTS-00663	2.4.1.2.3.3	2.4-8	Editorial correction	Add reference numbers as appropriate according to CTS-00666.	0
CTS-00664	2.4.1.2.3.3	2.4-8	Editorial correction	Delete "contributing".	0
CTS-00665	2.4.1.2.3.3	2.4-8	Update	Change "16,113 sq mi" to "25,679 sq mi".	0
CTS-00593	2.4.11.5	2.4-38	Editorial correction	Remove “to the cooling water system flow”.	0
CTS-00655	2.4.12.2.4	2.4-46	Editorial correction	Change “X” to “XX”.	0
CTS-00513	2.4.12.2.4 2.4.12.2.5	2.4-46 through	To reflect information	Re-write section reflecting RAI #1.	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_ 2.4.13-1 through RCOL2_ 2.4.13-7	2.4.12.3.1 2.4.12.5 2.4.13	2.4-64	provided during acceptance review		
CTS-00656	2.4.12.3.1	2.4-51	Editorial correction	Delete “(or are) expected to be”.	0
CTS-00657	2.4.12.3.1	2.4-52	Editorial correction	Change X to lower-case in mathematical expressions.	0
CTS-00658	2.4.12.5	2.4-53	Editorial correction	Add “aquifer”.	0
CTS-00659	2.4.13	2.4-56	Editorial correction	Change “Kd” to K_d”.	0
CTS-00666	2.4.16	2.4-63	Editorial correction	Add new references.	0
CTS-00589	Table 2.4.1-203	2.4-68 through 2.4-70	Erratum	Add reference citations.	0
CTS-00654	Table 2.4.1-203	2.4-68 through 2.4-70	Editorial correction	Change header titles and lower case from MSL to msl.	0
CTS-00655	Table 2.4.1-203	2.4-68 through 2.4-70	Erratum	Change values to match reference.	0
CTS-00588	Table 2.4.1-206	2.4-72	Erratum	Change “8186” to” 6354” and “0.383” to “0.362”. Add reference citations.	0
CTS-00594	2.5.1	2.5-53	Clarification	Add “potable” and “beneath the site”.	0
CTS-00599	2.5.2	2.5-61 2.5-62	Editorial correction	Delete the semi-colon in the bullet item list.	0
CTS-00595	2.5.2	2.5-61	Editorial correction	Remove IBR statement.	0
CTS-00515	2.5.2.5.1	2.5-110 through 2.5-113	To reflect information provided during acceptance review	Add three pages to clarify discussion.	0
CTS-00516	2.5.2.6.1.1 2.5.2.6.1.2	2.5-113 2.5-117	To reflect information provided during acceptance review	Revise Subsection reflecting commitment to NRC.	0
CTS-00667	2.5.4.3.3	2.5-166	Editorial correction	Change "The average elevation of the top of engineering Layer C is about 780 ft to 782 ft	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				below the Unit 3 power block, and about 782 ft to 784 ft below the Unit 4 power block (Figure 2.5.4-214)." to "The average elevation of the top of engineering Layer C is approximately 782 ft below the Unit 3 and Unit 4 power block (Figure 2.5.4-214)".	
CTS-00597	2.5.4	2.5-121	Editorial correction	Remove IBR statement.	0
CTS-00514	2.5.4.5.4	2.5-177 2.5-179	To reflect information provided during acceptance review	Revise Subsection reflecting commitment to NRC.	0
CTS-00517	2.5.4.8	2.5-187	To reflect information provided during acceptance review	Revise Subsection reflecting commitment to NRC.	0
CTS-00598	2.5.5	2.5-195	Editorial correction	Remove IBR statement.	0
CTS-00515	2.5.2.5	2.5-224	Editorial correction	Revise Subsection reflecting commitment to NRC.	0
CTS-00515	2.5.7	2.5-227 2.5-228	To reflect information provided during acceptance review	Add references 2.5-432 through 2.5-436	0
CTS-00515	2.5.7	2.5-228	To reflect information provided during acceptance review	Add reference 2.5-432.	0
CTS-00668	Table 2.5.1-201	2.5-229 2.5-230	Editorial correction	Delete "from the Studies of Madole (1988), Crone and Luza (1990), and Swan et al. (1993)" from the title of the table.	0
CTS-00669	Table 2.5.1-201	2.5-230	Editorial correction	Add reference citations.	0
CTS-00672	Table 2.5.1-202	2.5-231	Editorial correction	Delete notes.	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00673	Table 2.5.1-203	2.5-232	Editorial correction	Add reference citations.	0
CTS-00673	Table 2.5.1-203	2.5-232	Editorial correction	Delete and rewrite notes.	0
CTS-00670	Table 2.5.1-205	2.5-252	Editorial correction	Add reference citations.	0
CTS-00671	Table 2.5.1-206	2.5-254	Editorial correction	Add reference citations.	0
CTS-00674	Table 2.5.2-227	2.5-312	Editorial correction	Delete references in notes.	0
CTS-00515	List of Tables List of Figures	2-xxxii 2-xlviii	Commitment to NRC	Add Tables 2.5.2-230 through 2.5.2-235. Add Figures 2.5.2-240 through 2.5.2-246.	0
CTS-00516	List of Tables List of Figures	2-xxxii 2-xlviii	Commitment to NRC	Add Tables 2.5.2-236 and 2.5.2-237. Add Figures 2.5.2-247 through 2.5.2-252.	0
CTS-00515	Tables 2.5.2-230 through 2.5.2-237	-	To reflect information provided during acceptance review	Add new Tables.	0
CTS-00516	Figures 2.5.2-240 through 2.5.2-250	-	To reflect information provided during acceptance review	Add new Figures	0
MET-04	List of Tables	2-xxiv, 2-xxv	Erratum	Add “Dallas” in front of “Fort Worth” and “Airport” after “Fort Worth” for table number 2.3-296	1
CTS-00696	2.2.2.2.8	2.2-5	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Changed distance for DeCordova to 9.35 miles.	1
CTS-00697	2.2.2.6	2.2-8	Increase information as discussed with NRC during the 03-23-25-09	Added clarification that rail transport of hazardous materials is outside the 5 mile radius of CPNPP 3 & 4	1

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			Hazards Analysis Audit		
CTS-00699	2.2.2.7.1	2.2-9	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Added clarifying statement that the airports listed were predominant airports in the area outside 10 miles that did not exceed the 1000 D^2 criterion. Added back in the discussion for each predominant airport in the area outside the 10 miles.	1
CTS-00698	2.2.3.1.1.2	2.2-12	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Added clarifying discussion on how the Wolf Hollow hazardous materials were sceened for the hazards analysis since quantities were not made available.	1
CTS-00698	2.2.3.1.3.1	2.2-17	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Added clarifying discussion on how the Wolf Hollow hazardous materials were sceened for the control room habitability analysis since quantities were not made available.	1
CTS-00696	2.2.3.1.3.2.2	2.2-18	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Clarified discussion regarding DeCordova was analyzed for Hazards and Control Room Habitablilty analyses even though the distance is outside the 5 mile radius of Units 3 & 4.	1
CTS-00698	Table 2.2-205	2.2-32	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Added footnote that the quantities of chemicals were not made available for Wolf Hollow and a pointer added to indicate what sections have the sceening criteria utilized for Wolf Hollow.	1

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00696	Table 2.2-214	2.2-43	Increase information as discussed with NRC during the 03-23-25-09 Hazards Analysis Audit	Added IDLH and Max concentration in Control Room and footnote (b) indicating that DeCordova was conservatively analyzed even though it is outside the 5 mile radius of U3/4. Distance to nearest Units 3 and 4 MCR Inlet for DeCordova SES has been revised from 3.6 to 3.7.	1
CTS-00696	Figure 2.2-201		Erratum	Corrected the figure since the location of DeCordova, which is outside the 5 mile radius of CPNPP Units 3 & 4, showed DeCordova inside the 5 mile radius	1
MET-03	2.3.1.2.4	2.3-14	Increase information as discussed with the NRC.	Add “16” to number of days each year; remove “monthly and regional” and add “by county” to wind events to reconcile thunderstorm information.	1
MET-04	2.3.1.2.8	2.3-20	Erratum	Add “the” in front of Dallas Fort Worth Airport	1
MET-13	2.3.2.1.2	2.3-22	Erratum	Replace “2001 through 2006” with “2001 – 2004 and 2006” to describe which data years were used.	1
MET-13	2.3..2.1.3	2.3-27	Erratum	Replace “2001- 2006” with “2001 – 2004 and 2006” to describe which data years were used.	1
MET-04	2.3.2.1.4	2.3-27	Erratum	Add “Dallas” in front of “Fort Worth”	1
MET-13	2.3.2.2.4	2.3-32	Erratum	Add “Fort” for the years “2001 – 2006”	1
MET-3 MET-13	Table 2.3-211	2.3-83	Erratum	Replace numbers in column “Average per Yr (#/yr) and Replace “2006 and (-24 yr) with	1

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				“7/31/2006”	
MET-13	Table 2.3-285	2.3-164	Errata	Replace “2001 – 2006” with “2001 – 2004 and 2006” to describe which data years were used.	1
MET-04	Table 2.3-286	2.3-165	Erratum	Add “Dallas” in front of “Fort Worth” for the title.	1
MET-04	Table 2.3-296	2.3-177	Erratum	Add “Dallas” in front of Fort Worth and “Airport” after Worth in the title	1
MET-04	Table 2.3-299	2.3-180 2.3-181	Erratum	Add “Dallas” in front of ”Fort Worth” in the title	1
CTS-00554	List of Tables	2-xxxiii	Increase information as discussed with the NRC to summarize the reports provided in Luminant’s letter TXNB-08027 to NRC dated November 4, 2008.	Added Tables 2.5.4-228 through 2.5.4-231	2
CTS-00554	List of Figures	2-I	Increase information as discussed with the NRC to summarize the reports provided in Luminant’s letter TXNB-08027 to NRC dated November 4, 2008.	Added Figure 2.5.4-245	2
CTS-00703	Table 2.3-332	2.3-233 2.3-234	To reflect CPNPP Units 3 and 4 compliance with RG 1.23	Added “Second Proposed Revision, April 1986” to the footnotes	2
CTS-00554	2.5.4.10.1	2.5-189	Increase information as discussed with the NRC to summarize the reports provided in Luminant’s letter TXNB-08027 to NRC dated November	Additional discussion and equations to reflect what calculations and analyses were performed to demonstrate bearing capacity.	2

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			4, 2008.		
CTS-00554	2.5.4.10.2	2.5-190	Increase information as discussed with the NRC to summarize the reports provided in Luminant's letter TXNB-08027 to NRC dated November 4, 2008.	Additional discussion on settlement, including calculations, equations and discussion of laboratory test results, layered versus unlayered method.	2
CTS-00554	2.5.4.10.3	2.5-191	Increase information as discussed with the NRC to summarize the reports provided in Luminant's letter TXNB-08027 to NRC dated November 4, 2008.	Additional information added to excavation rebound potential.	2
CTS-00554	2.5.7	2.5-228	Increase information as discussed with the NRC to summarize the reports provided in Luminant's letter TXNB-08027 to NRC dated November 4, 2008.	Added references 2.5-432 through 2.5-434 to reflect additional discussion on bearing capacity and settlement subsection discussed.	2
CTS-00554	Tables 2.5-4-228 through 2.5.4-231	-	Increase information as discussed with the NRC to summarize the reports provided in Luminant's letter TXNB-08027 to NRC dated November 4, 2008.	Added new tables to reflect bearing capacity discussion and settlement discussion within subsections.	2
CTS-00554	Figure 2.5.4-245		Increase information as discussed with the NRC to	Added Figure 2.5.4-245.	2

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			summarize the reports provided in Luminant's letter TXNB-08027 to NRC dated November 4, 2008.		
HYDSV-23	List of Figures	2xliv	Hydrology Site Safety Visist	Added figures to show flow paths to SCR.	4
HYDSV-06 HYDSV-07	Table 2.0-1R		Hydrology Site Safety Visit	Changed the maximum flood level.	4
HYDSV-04	2.4.1.2	2.4-4	Hydrology Site Safety Visit	Clarified what portions of the Brazos River basin were chosen for the dam failure safety analysis.	4
HYDSV-05	2.4.1.2	2.4-5	Hydrology Site Safety Visit	Updated section to reflect what reservoirs were considered in the dam failure safety analysis.	4
HYDSV-02	2.4.2.1	2.4-12 2.4-13	Hydrology Site Safety Visit	Added maximum flood level and design basis flood elevation.	4
HYDSV-14	2.4.2.2	2.4-13 2.4-14	Hydrology Site Safety Visit	Changed water surface elevation for flood design.	4
HYDSV-06 HYDSV-07	2.4.2.3	2.4-16	Hydrology Site Safety Visit	Changed the tail water elevation.	4
HYDSV-06 HYDSV-07	2.4.3	2.4-18	Hydrology Site Safety Visit	Revised the surface water elevation for the probably maximum flood.	4
HYDSV-06 HYDSV-07	2.4.3.1	2.4-19	Hydrology Site Safety Visit	Revised the critical temporal distribution for the probably maximum precipitation.	4
HYDSV-06 HYDSV-07	2.4.3.3	2.4-20 2.4-21	Hydrology Site Safety Visit	Added discussion justifying the use of the Snyder's hydrograph applicability under PMF conditions and added a storage discharge relationship was linearly extrapolated to account	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				for discharge from elevation 791 ft msl to 795 ft. msl.	
HYDSV-06 HYDSV-07	2.4.3.4	2.4-22	Hydrology Site Safety Visit	Changed the SCR peak flood volumetric flow rate.	4
HYDSV-06 HYDSV-07	2.4.3.5	2.4-22	Hydrology Site Safety Visit	Changed the surface water elevation for the HEC-HMS and HEC-RAS models.	4
HYDSV-06 HYDSV-07	2.4.3.6	2.4-22 2.4-23	Hydrology Site Safety Visit	Revised the critical fetch length, critical duration wind speed, wave height, runup, maximum wind speed, and setup for the dam failure analysis.	4
HYDSV-04	2.4.4	2.4-24	Hydrology Site Safety Visit	Clarified assumptions of what dam failures were used in the dam failure analysis and why.	4
HYDSV-09	2.4.4.1	2.4-27	Hydrology Site Safety Visit	Clarified which reservoirs in the Brazos River Basin where used in the flooding analysis. Added discussion of what volumes of reservoir water were used in the dam failure analysis. Changed the maximum surface water elevation.	4
CTS-00817 HYDSV-10 HYDSV-11	2.4.5	2.4-29	Hydrology Site Safety Visit	Edited 5th paragraph 2nd to last sentence of section from "Any effects on the Squaw Creek.. to read "Any effects on SCR…". Added discussion as to why the seismic induced wave and the landslide induced wave is not plausible for SCR. Changed the water surface elevation due to wind activity and changed the PMF coincident wind wave.	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
HYDSV-03	2.4.5	2.4-29	Hydrology Site Safety Visit	Clarified that the plant grade elevation is at 822 ft msl.	4
HYDSV-12 HYDSV-13	2.4.6	2.4-30	Hydrology Site Safety Visit	Added discussion that landslide and seismic induced waves are note plausible for SCR.	4
HYDSV-14	2.4.7	2.4-32	Hydrology Site Safety Visit	Changed the maximum flood elevation. Added a discussion regarding the maximum potential ice thickness and that freezing protection was provided for the ESWS cooling towers and ESW Pump House.	4
HYDSV-16	2.4.11.5	2.4-38	Hydrology Site Safety Visit	Added a discussion regarding the control of the ESWS and CWS cooling towers with makeup flow rates.	4
HYDSV-20	2.4.12.2.4	2.4-46 2.4-47	Hydrology Site Safety Visit	Updated the Groundwater Level Fluctuations to include the 2008 precipitation data and the resulting effect on the groundwater level fluctuations results.	4
HYDSV-20	2.4.12.2.4	2.4-46 2.4-47	Hydrology Site Safety Visit	Removed previous RCOL2_2.4.4.13-4 addition of “undifferentiated fill/regolith and” as well as, “indicating perched groundwater at these locations.”	4
HYDSV-18 HYDSV-24	2.4.12.2.5.1	2.4-49	Hydrology Site Safety Visit	Revised to clarify the conservatism used in porosity to calculate liquid effluent travel times.	4
HYDSV-23	2.4.12.3.1	2.4-51	Hydrology Site Safety Visit	Revised section to describe the post-construction movement o groundwater to support the liquid effluent release model provided in Section	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				2.4.13.	
HYDSV-26	2.4.12.4	2.4-53	Hydrology Site Safety Visit	Revised to reflect that a groundwater monitoring program will be developed before fuel load.	4
CTS-00808 HYDSV-30	2.4.13	2.4-54	Hydrology Site Safety Visit	Corrected Figure typo to 2.4.12-209. Discussed the alternate conceptual model and added a reference to new Figures 2.4.12-212-214.	4
HYDSV-28	2.4.13.1	2.4-55	Hydrology Site Safety Visit	Clarified conclusion that no chemical agents could have an effect on the transport characteristics of the liquid effluent.	4
HYDSV-30	2.4.13.2	2.4-55	Hydrology Site Safety Visit	Added clarification regarding the alternate pathways chosen and introduced new Figures 2.4-12-212 through 2.4.12-214 showing the new pathways and cross sections and discussed the hydraulic gradient figures showing the reason why GW movement SE and SW are not plausible release pathways.	4
HYDSV-17 HYDSV-19 HYDSV-23 HYDSV-30	2.4.13.2	2.4-55	Hydrology Site Safety Visit	Added paragraph to introduce new cross section figures and pathway figure.	4
HYDSV-17 HYDSV-19 HYDSV-23 HYDSV-30	2.4.13.2	2.4-55	Hydrology Site Safety Visit	Added two more bullets on what alternate conceptual model parameters were used in developing the site conceptual model plausible pathways.	4
HYDSV-17 HYDSV-19 HYDSV-23 HYDSV-30	2.4.13.3	2.4-55	Hydrology Site Safety Visit	Added a discussion that rainfall infiltration is not a contributing factor that would affect the liquid effluent release analysis.	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
HYDSV-29 HYDSV-31	2.4.13.4	2.4-55	Hydrology Site Safety Visit	Corrected the distances to the nearest water supply wells both in the Glen Rose formation and the Twin Mountains formation.	4
HYDSV-17 HYDSV-19 HYDSV-29 HYDSV-31	2.4.13.4	2.4-61	Hydrology Site Safety Visit	Added a clarification as to why the vertical release pathway is not plausible based upon the Unit 1 and 2 study previously performed.	4
HYDSV-23	2.4.13.4	2.4-61	Hydrology Site Safety Visit	Added reference to new Cross Section figures and pathway Figures 2.4-12-212 through 2.4.12-214.	4
HYDSV-17 HYDSV-19 HYDSV-23 HYDSV-30	2.4.13.5	2.4-55	Hydrology Site Safety Visit	Revised to discuss four release pathways. Revised to include discussion of why alternate pathways moving SE or SW from Units 3 or 4 would not be plausible.	4
HYDSV-17 HYDSV-23 HYDSV-30	2.4.13.5	2.4-55	Hydrology Site Safety Visit	Changed to plausible pathways 3a, 3b, 4a, 4b and changed travel times to SCR, and deleted current pathways. Changed travel times and identified the shortest travel time to SCR. Referred to cross section figures and new pathways.	4
HYDSV-17 HYDSV-23 HYDSV-30	2.4.13.7	2.4-55	Hydrology Site Safety Visit	Revised base mat elevation for A/B and specified subsection for site specific hydrogeologic data and core boring stratigraphy for A/B.	4
HYDSV-17 HYDSV-23 HYDSV-30	2.4.13.7	2.4-55	Hydrology Site Safety Visit	Changed travel times for the new pathways, specified what subsection discusses the comparison of U1/2	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				vertical pathway study, and made minor editorials.	
HYDSV-05	References 2.4-269 and 2.4-270	2.4-63	Hydrology Site Safety Visit	Added two new references to describe potential reservoir sites considered in the dam failure analysis.	4
HYDSV-15	References 2.4-271 and 2.4-272	2.4-63	Hydrology Site Safety Visit	Added two new references for the ice effects analysis Section 2.4.7.	4
HYDSV-02	Table 2.4.2-204	2.4-87	Hydrology Site Safety Visit	Added the datum elevation for footnote b.	4
HYDSV-06 HDYSV-07	Table 2.4.2-208	2.4-91	Hydrology Site Safety Visit	Changed the tail water elevation.	4
HYDSV-06 HYDSV-07	Table 2.4.3-202	2.4-93	Hydrology Site Safety Visit	Changed the PMP degree storm orientation.	4
HYDSV-06 HYDSV-07	Table 2.4.3-207	2.4-102	Hydrology Site Safety Visit	Changed the watershed sub-basin characteristics.	4
HYDSV-23	Table 2.4.12-211	2.4-149 through 2.4-152	Hydrology Site Safety Visit	Replaced Groundwater and Velocity Times Based Upon Post-Construction Configuration.	4
HYDSV-02	Figures 2.4.2-201 2.4.2-202 2.4.3-202 2.4.3-209 2.4.4-201 2.4.4-202	--	Hydrology Site Safety Visit	Added horizontal and vertical datums; added additional fetches; clarified watershed boundaries; and added datum sources.	4
HYDSV-20	Figure 2.4.12-209	--	Hydrology Site Safety Visit	Replaced the hydrographs for monitoring wells with expanded scale and precipitation data.	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
HYDSV-23	Figures 2.4.12-212 2.4.12-213 2.4.12-214	--	Hydrology Site Safety Visit	Added new Figures for Groundwater Flow Paths for Liquid Effluent Release and Cross Sections	4
RCOL2_02.05.02-07	2.5.2.5	2.5-110	Response to RAI No. 11 Luminant Letter no.TXNB-09035 Date 8/28/2009	Changed 6000 ft/sec to 5800 ft/sec.	-
RCOL2_02.05.02-21	2.5.2.2.1.1	2.5-73	Response to RAI No. 11 Luminant Letter no.TXNB-09035 Date 8/28/2009	Changed Figure 2.5.2-233 to Figure 2.5.2-203.	-
RCOL2_02.05.02-21	2.5.2.4.2.3.2.1	2.5-96-2.5.-97	Response to RAI No. 11 Luminant Letter no.TXNB-09035 Date 8/28/2009	Changed Figure 2.5.-211 to Figure 2.5.1-211.	-
RCOL2_02.05.02-21	Table 2.5.2-208 2.5.2-209	2.5-286 2.5-287	Response to RAI No. 11 Luminant Letter no.TXNB-09035 Date 8/28/2009	Changed data collection date on Table 2.5.2-208 from 2008 to 2007.	-
RCOL2_02.05.02-21	Table 2.5.2-220	2.5-300	Response to RAI No. 11 Luminant Letter no.TXNB-09035 Date 8/28/2009	Added shaded cells in Table 2.5.2-220.	-
RCOL2_02.05.04-11	2.5.4.5.4.1.2	2.5-179 2.5-228	Response to RAI No. 22 Luminant Letter no.TXNB-09035 Date 8/28/2009	Revised subsection for RAI response.	-

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_02.05.04-11	2.5.4.5.4.1.2	2.5-243	Response to RAI No. 22 Luminant Letter no.TXNB-09035 Date 8/28/2009	Added references for RAI response.	-
RCOL2_02.05.04-12	2.5.4.5.4.1.2	2.5-179 2.5-228	Response to RAI No. 22 Luminant Letter no.TXNB-09035 Date 8/28/2009	Revised subsection for RAI response.	-
RCOL2_02.05.04-12	2.5.7	2.5-243	Response to RAI No. 22 Luminant Letter no.TXNB-09035 Date 8/28/2009	Added references for RAI response.	-
RCOL2_02.05.01-05	2.5.1.1.3.1	2.5-10	Response to RAI No. 21 Luminant Letter no.TXNB-09035 Date 8/28/2009	Changed southeastern to southwestern.	-
RCOL2_02.05.01-01	2.5.1.1.3.1 2.5.1.1.3.2	2.5-11 2.5-12	Response to RAI No. 14 Luminant Letter no.TXNB-09035 Date 8/28/2009	Revised subsection for RAI response.	-

Chapter 3

Chapter 3 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00638	3.3.1.2	3.3-1	Clarification	Add “CPNPP Units 3 and 4 do not have site-specific seismic category II buildings and structures”.	0
CTS-00600	3.7.1	3.7-3	Editorial correction	Change “is” to “has been”.	0
MAP-03-001	3.7.4.2 3.7.5	3.7-12 3.7-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.7(15)	0
MAP-03-002	3.7.4.5 3.7.5	3.7-12 3.7-13 3.7-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.7(18)	0
CTS-00532	Table 3.7.2-1R	3.7-17 3.7-18	Editorial correction	Revise LMN to highlight changes.	0
MAP-03-003	3.8.1.4.1.3 3.8.6	3.8-1 3.8-13 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(1)	0
MAP-03-004	3.8.1.5.1.2 3.8.1.5.2.2 3.8.6	3.8-1 3.8-1 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(2)	0
CTS-00602	3.8.1	3.8-2	Clarification	Change “Chapter 2” to “Subsection 2.5.4”.	0
MAP-03-005	3.8.1.6 3.8.6	3.8-2 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(4)	0
MAP-03-006	3.8.1.6 3.8.6	3.8-2 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(5)	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
MAP-03-007	3.8.1.6 3.8.6	3.8-2 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(6)	0
MAP-03-008	3.8.1.6 3.8.6	3.8-3 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(8)	0
MAP-03-009	3.8.1.6 3.8.6	3.8-3 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(9)	0
MAP-03-010	3.8.1.6 3.8.6	3.8-3 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(12)	0
MAP-03-011	3.8.1.6 3.8.6	3.8-3 3.8-14	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.8(13)	0
CTS-00607	3.8.4.1.3.2	3.8-6 3.8-7	Editorial correction	Change “the ESW pump houses” to “UHS ESW pump house”.	0
MAP-03-012	3.8.4.7	3.8-11	Revision of COL 3.8(22) Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Change “Monitoring of seismic category I structures is required to be performed” to “a site-specific program for monitoring and maintenance of seismic category I structures is performed”.	0
CTS-00603	Table 3.9-202	3.8-18	Consistent with DCD Rev.1	Change unit and number in the table.	0
CTS-00604	3.9.3.4.2.5	3.9-2	Editorial correction	Clarify wording.	0
CTS-00531	3.9.3.4.2.5	3.9-2	Editorial correction	Change “are” to “is”.	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00605	Table 3.9-201	3.9-5	Editorial correction	Change COL item number.	0
MAP-03-014	3.10 3.10.7	3.10-1 3.10-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.10(10)	0
CTS-00606	3.11	3.11-1	Clarification	Replace EQ program implementation dates with milestones.	0
CTS-00639	3.11.5	3.11.3	Editorial correction	Change “Table 3D-201 by completion of [Later]” to “the Equipment EQ Technical Report (Reference 3.11.3)”.	0
MAP-03-015	3.13.1.2.3 3.13.3	3.13-1 3.13-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.13(1)	0
MAP-03-016	3.13.1.2.5 3.13.3	3.13-1 3.13-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 3.13(2)	0
DCD_3.5.1.1-04	3.5	3.5-1 3.5-4	Reflect Response to DCD RAI No. 127	Change section number and title	3
RCOL2_03.05.01.03-1	3.5.1.3.2	3.5-2	Response to RAI No. 12 Luminant Letter no.TXNB-09033 Date 08/24/2009	Inserted a description of turbine valve test frequency.	-
RCOL2_10.04.08-1	Table 3.2-201	3.2-5	Response to RAI No. 17 Luminant Letter no.TXNB-09034 Date 08/24/2009	For Item #4 under the “System and Components” column for the Startup steam generator (SG) blowdown system, correct the information for the Equipment Class, location, Quality Group, Codes and Standards, and Seismic	-

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				Category. In addition, modify Note 1.	
DCD_03.02.01-6	3.2.1	3.2-5	Reflect Response to DCD RAI No. 287	Change the description of note and add note.	4
CTS-00804	3.2.1	3.2-5	Editorial correction	Left-justify first column	4

Chapter 4

Chapter 4 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
MAP_4.4.7-2	4.4	4.4-1	To be consistent with next DCD revision (Rev.2)	Delete COL 4.4 (1) and associated description	4

Chapter 5

Chapter 5 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00528	5.2.1.2	5.2-1	Editorial correction	Include words about RG 1.84.	0
CTS-00675	5.2.1.2	5.2-1	Editorial correction	Add "Units 3 and 4" after Comanche Peak Nuclear Power Plant. Delete a period in LMN	0
RCOL2_05.03-1	5.3.2.3	5.3-3	Responses to RAI No. 2 Luminant Letter TXNB-09010 Dated 5/1/2009	Add clarification about the timing of submitting PTS evaluation using the as–procured reactor vessel material properties.	-
RCOL2_05.0 3.02-2	5.3.2.1	5.3-2	Response to RAI No. 8 Luminant Letter no.TXNB-09028 Date 8/7/2009	Include a commitment to update P/T limits before fuel load. The RAI No.2 change is superseded by RAI No. 8.	-
DCD_05.03. 02-1	5.3.2.1	5.3-2	Reflect Response to DCD RAI No. 287	Stated that generic PTLR will be applied for CPNPP 3&4.	4

Chapter 6

Chapter 6 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00518 CTS-00644	6.4.4	6-i 6.4-1 6.4-3 1.8-43	To reflect resolution of acceptance review issue	Include dose evaluation in the control room due to a post-accident release from the other US-APWR unit or existing CPNPP unit.	0
	6.4.4		Editorial correction	Add Subsection “6.4.4.2” in Table 1.8-201 and Subsection 6.4.7.	0
CTS-00642	6.1	6.1-1	Update	All 6.1 COL Items have been deleted from the DCD. This FSAR section is now IBR with no departures or supplements.	0
MAP-06-001	6.1.1.2.2	6.1-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.1(1)	0
MAP-06-002	6.1.1.1	6.1-1 6.1-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.1(2)	0
MAP-06-003	6.1.1.2.1	6.1-1 6.1-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.1(3)	0
MAP-06-004	6.1.1.2.1	6.1-1 6.1-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.1(4)	0
MAP-06-005	6.1.2	6.1-2 6.1-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.1(5)	0
MAP-06-006	6.2.1.1.3.4 6.2.1.5.7	6.2-1 6.2-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.2(1)	0
MAP-06-007	6.2.2.3 Table 6.2.2-2R	6.2-1 6.2-4 6.2-6	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.2(9)	0
MAP-06-008	6.2.4.2	6.2-2 6.2-3	Deletion of COL item. Letter MHI Ref:UAP-	Delete COL 6.2(6)	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			HF-08259, dated on Nov.7, 2008		
MAP-06-009	6.2.5.2	6.2-2 6.2-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.2(7)	0
DCD_06.02.06-2	6.2.6.1	6.2-3	DCD_RAI 06.02.06-2	Change “first sentence “ to “first and second sentences”.	0
CTS-00643	6.3	6.3-1	Update	All 6.3 COL Items have been deleted from the DCD. This FSAR section is now IBR with no departures or supplements.	0
MAP-06-011	6.3.2.8	6.3-1 6.3-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.3(3)	0
MAP-06-012	6.3.2.2.4	6.3-1 6.3-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.3(4)	0
MAP-06-013	6.3.2.4	6.3-1 6.3-2	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.3(6)	0
MAP-06-014	6.4.3 6.4.7	6.4-1 6.4-3	Revision of COL 6.4(2)	Revise COL Item to only discuss automatic actions and manual procedures for the MCR HVAC system in the event of postulated toxic gas release.	0
MAP-06-015	6.4.2.2.1	6.4-1 6.4-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.4(4)	0
CTS-00652	6.4.4.2 6.4.7	6.4-2 6.4-3	Re-evaluation of COL Item	Associate COL 6.4(2) with Subsection 6.4.4.2.	0
CTS-00653	6.4.4.2	6.4-3	Erratum	Change “5.2 ppm “ to “5.7 ppm”.	0
MAP-06-016	6.5.1.7	6.5-1	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 6.5(4)	0

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
MAP-06-018	6.6.8	6.6-1	Revision of COL 6.6(2)	Revise description to only identify the implementation milestone of the program.	0
CTS-00696	6.4.4.2	6.4-1	NRC Staff Reviewer Comment Incorporation from 03-23-25-09 Hazards Analysis Audit	Added pointer to Table 2.2-214 for toxic chemicals that do not meet RG 1.78 screening criteria.	1
DCD_06.01.02-1	6.1	6.1-1	Reflect Response to DCD RAI No. 365 revision 1	Added COL 6.1(7) coating program	4

Chapter 7

Chapter 7 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_7.04_1	7.4.1.6	7.4-1	Response to RAI No.4 Luminant Letter no.TXNB-09020 Date 5/26/2009	Add a description of reference; FSAR subsection 9.2.5.	-
RCOL2_7.05_1	7.5.1.6.2	7.5-1	Response to RAI No.5 Luminant Letter no. TXNB-09020 Date 5/26/2009	Revise the description of EOF capability. EOF has identical information as TSC and MCR, but does not control capability.	-
CTS-00721	Table 7.4-201	7.4-2	Editorial correction	Change the Safe shutdown column of ESWS from “No” to “Yes”.	4
DCD_07.05-17	7.5.1.1 7.5.4 Table 7.5-201	7.5-1 7.5-2 7.5-3	Reflect Response to DCD RAI No. 238	The descriptions of Site-specific type E PAM variables for metrological parameters are added.	4

Chapter 8

Chapter 8 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00451	List of Figures, Figure 8.2-201	8-iii 8.2-23	Editorial correction	Add “Relevant Portions of” to the title of the Figure 8.2-201.	0
CTS-00640	8.2.1.2	8.2-3	Editorial correction	Change “Any” to “Both of any”.	0
CTS-00686	8.2.1.2.1.1	8.2-5	Editorial correction	Delete “from”.	0
CTS-00641	8.2.1.2.1.1	8.2-6	Erratum	Change “is” to “are”.	0
CTS-00477	8.2	8.2-6	Clarification	Change description of offsite power system.	0
CTS-00479	8.4	8.4-1	Editorial correction	Change section title in bold font.	0
CTS-00722	8.3	8.3-2	COL item closure of the original COL Holder Items	Change the description of Grounding and Lightning Protection System design information.	4

Chapter 9

Chapter 9 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00586	9.2.1.2.1	9.2-1 9.2-2	Consistent with Subsection 9.4.5.2.6	Change “ESWP house” to “UHS ESW pump house”.	0
CTS-00608	9.4	9.4-7	Erratum	Change heating coil capacity of EFP (M/D) Area Air Handling Unit from “1 kW” to “2 kW”.	0
DCD_09.05.01-6	9.5.1.3 9.5.9	9.5-3 9.5-18	DCD_RAI 09.05.01-6	Add Subsection 9.5.1.3.	0
DCD_09.05.01-15	Table 9.5.1-1R	9.5-46	DCD_RAI 09.05.01-15	Add LMNs in Table 9.5.1-1R and Table 9.5.1.2R.	0
DCD_09.05.01-7	Table 9.5.1-1R	9.5-55	DCD_RAI 09.05.01-7	Add “see Subsection 9.5.1.3” to Table 9.5.1.1R.	0
DCD_09.05.01-5	Table 9.5.1-1R	9.5-56	DCD_RAI 09.05.01-5	Fill in Remarks on Table 9.5.1-1R.	0
DCD_09.05.01-15	Table 9.5.1-2R	9.5-112 9.5-113	DCD_RAI 09.05.01-15	Add LMNs in Table 9.5.1-1R and Table 9.5.1.2R.	0
DCD_09.02.04-1	9.2.10	9.2-12	Reflect Response to DCD RAI No. 125	Revised text in CP COL 9.2(10) for clarity.	3
DCD_09.02.04-2	9.2.10	9.2-13	Reflect Response to DCD RAI No. 125	Revised text in CP COL 9.2(16) for clarity.	3
DCD_09.02.01-4	9.2.1.2.1	9.2-1	Reflect Response to DCD RAI No. 326-2279, Question 4	Add a paragraph to CP COL 9.2(7) to define boundary between safety-related and non-safety-related boundary of the ESW as the vent and drain valves of the strainers and heat exchangers	5
DCD_09.02.01-17	9.2.1.2.1	9.2-1	Reflect Response to DCD RAI 326-2279, Question 17	Add CP COL 9.2(26) to identify maintenance and test procedures to monitor and flush out debris shall be implemented.	5
DCD_09.02.01-30	9.2.1.2.1	9.2-1	Reflect Response to DCD RAI 326-2279, Question 30	Add CP COL 9.2(25) to clarify proper filling and venting procedures to prevent water hammer.	5

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
DCD_09.02.01-30	9.2.1.3	9.2-2	Reflect Response to DCD RAI 326-2279, Question 30	Add second paragraph to COL 9.2(1) description of recovery procedures in the event that the UHS approaches low water level.	5
DCD_09.02.01-30	9.2.10	9.2-11	Reflect Response to DCD RAI 326-2279, Question 30	Add at the end of CP COL 9.2(1) “and recovery procedures when UHS approaches low water level.”	5
DCD_09.02.01-30	9.2.12	9.2-12	Reflect Response to DCD RAI 326-2279, Question 30	Revise CP COL 9.2(8) to read “The specific ESW chemistry requirements”	5
DCD_09.02.01-12,13,14,30	9.2.10	9.2-14	Reflect Response to DCD RAI 326-2279, Question 12,13,14 and 30	Add 9.2(25) The operating and maintenance procedures to address water hammer issues. This COL item is addressed in Subsections 9.2.1.2.1 and 13.5.2.1.	5
DCD_09.02.01-17	9.2.10	9.2-14	Reflect Response to DCD RAI 326-2279, Question 17	Add 9.2(26) Maintenance and test procedures to monitor and flush out debris. This COL item is addressed in Subsections 9.2.1.2.1 and 13.5.2.1	5

9.2 WATER SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.2.1.2.1 General Description

CP COL 9.2(7) Replace the first sentence of the first paragraph in DCD Subsection 9.2.1.2.1 with the following.

Figure 9.2.1-1R shows the piping and instrumentation diagrams (P&IDs) of the essential service water system (ESWS).

CP COL 9.2(25) Replace the eighth paragraph in DCD 9.2.1.2.1 with the following: DCD_09.02.01-12

Proper filling and venting procedures are followed to minimize the occurrence of water hammer and mitigate its effects. These are included in the Operating and Maintenance Procedures mentioned in Subsection 13.5.2.1 DCD_09.02.01-30

CP COL 9.2(8) Replace the sixth paragraph in DCD Subsection 9.2.1.2.1 with the following.

Chemicals are added to the basin to control corrosion, scaling, and biological growth. The water chemistry is managed through a Chemistry Control Program such as following a standard Langelier Saturation Index. The chemical injection system is described in Subsection 10.4.5.

CP COL 9.2(7) Replace the seventh paragraph in DCD Subsection 9.2.1.2.1 with the following.

The non-safety-related portion of the ESWS begins at the discharge side of the strainer and CCW heat exchangers vent and drain valves. The positions of these valves are controlled by the Operating and Maintenance Procedures mentioned in Subsection 13.5.2.1 in order to maintain water-tight conditions and prevent inadvertent draining of the ESW. DCD_09.02.01-4

Blowdown is used to maintain acceptable water chemistry composition. This is accomplished by tapping each essential service water pump (ESWP) discharge header. Additional description about blowdown is discussed in Subsection 9.2.5.

CP COL 9.2(26) Replace the fourteenth paragraph in DCD 9.2.1.2.1 with the following: DCD_09.02.01-17

Maintenance and test procedures (see Operating and Maintenance Procedures in Subsection 13.5.2.1) are followed to monitor and flush debris accumulated in the system.

CP COL 9.5(2) Add the following text after the last paragraph in DCD Subsection 9.2.1.2.1.

Each of the essential service water (ESW) lines in the reactor building (R/B) and in the ~~ESWP~~UHS ESW pump house is tapped to supply water to the fire protection water supply system (FSS), if required, after the safe-shutdown earthquake (SSE). Manually operated locked closed valves are provided in each of the tapped connections to draw water for the FSS. CTS-00586

9.2.1.2.2 Component Description

CP COL 9.2(6) Replace the sentence in DCD Subsection 9.2.1.2.2 with the following.

Table 9.2.1-1R shows the design parameters of the major components in the system.

9.2.1.2.2.1 ESWPs

CP COL 9.2(6) Replace the second sentence of the third paragraph in DCD Subsection 9.2.1.2.2.1 with the following.

Total dynamic head of the ESWP is 220 feet. Available net positive suction head (NPSH) with the lowest expected water level (after 30 days of accident mitigation) in the basin is approximately 40 feet.

9.2.1.3 Safety Evaluation

CP COL 9.2(1) Replace the eleventh paragraph in DCD Subsection 9.2.1.3 with the following.

Design of the basin provides adequate submergence of the pumps to assure the NPSH for the pumps. The basin is divided into two levels. One is approximately 12 feet lower than the other, and directly above it is installed the ESWP. The ESWP is designed to operate with the lowest expected water level (after 30 days of accident mitigation). The basins have sufficient water inventory to assure adequate cooling and NPSH for 30 days without makeup. This is discussed further in Subsection 9.2.5.

Recovery procedures contained in the Operating and Maintenance Procedures (see Subsection 13.5.2.1) are implemented if the UHS approaches low water level. DCD_09.02.01-30

CP COL 9.2(2) Replace the twelfth paragraph in DCD Subsection 9.2.1.3 with the following.

The lowest ambient temperature anticipated at the site does not result in the freezing of the ESW in the basin or the piping for the following reasons:

Manholes, handholes, inspection ports, ladder, and platforms are provided, as required, for periodic inspection of system components.

9.2.5.5 Instrumentation Requirements

CP COL 9.2(24) Replace the sentence in DCD Subsection 9.2.5.5 with the following.

Water level in each of the basins is controlled by level instrumentation that opens or closes the automatic valves in the makeup lines.

Two level transmitters and associated signal processors are provided for each basin to indicate water level in the basin and annunciate in the MCR for both the high and low water levels in the basin.

A water level signal at six inches below the normal water level causes the makeup water control valve to open. A signal at normal water level then causes the makeup control valve to close. A low level alarm annunciates in the MCR whenever the water level falls one foot below the normal water level.

During accident condition, level indications from the operating basins are used to alert the MCR operator to start the UHS transfer pump to transfer water from the idle basin to the operating basins.

Blowdown rate is controlled manually. The blowdown control valves close automatically upon receipt of a low water level signal or emergency core cooling system actuation signal. The valve is designed to fail in the close position. Failure of the valve to close is indicated in the MCR.

The conductivity cells are provided at the ESW pump discharge line and conductivity are indicated in the MCR.

Temperature elements are provided in each basin and temperatures are indicated in the MCR.

Local flow rate and pressure indicators located in each UHS transfer pump discharge header are used for pump performance testing.

The cooling tower fan is equipped with vibration sensors that alarm in the control room in the event of high vibration.

9.2.10 Combined License Information

Replace the content of DCD Subsection 9.2.10 with the following.

CP COL9.2(1) ***9.2(1)*** *The evaluation of ESWP at the lowest probable water level of the UHS <u>and the recovery procedures when UHS approaches low water level</u>* DCD_09.01.02-30

This COL item is addressed in Subsection 9.2.1.3.

CP COL 9.2(2) ***9.2(2)*** *The protection against adverse environmental, operating and accident condition that can occur such as freezing, thermal over pressurization*

This COL item is addressed in Subsection 9.2.1.3.

CP COL 9.2(3) ***9.2(3)*** *Source and location of the UHS*

This COL item is addressed in Subsection 9.2.5.2.

CP COL 9.2(4) ***9.2(4)*** *The location and design of the ESW intake structure*

This COL item is addressed in Subsection 9.2.5.2.

CP COL 9.2(5) ***9.2(5)*** *The location and the design of the discharge structure*

This COL item is addressed in Subsection 9.2.5.2.

CP COL 9.2(6) ***9.2(6)*** *The ESWP design details – required total dynamic head, NPSH available*

This COL item is addressed in Subsection 9.2.1.2.2, 9.2.1.2.2.1 and Table 9.2.1-1R.

CP COL 9.2(7) ***9.2(7)*** *The design of ESWS related with the site specific UHS*

This COL item is addressed in Subsections 9.2.1.2.1, 9.2.1.3, 9.2.1.5.4 and Figure 9.2.1-1R.

CP COL 9.2(8) ***9.2(8)*** *The ESW specific chemistry requirements* DCD_09.02.01-30

This COL item is addressed in Subsection 9.2.1.2.1.

CP COL 9.2(9) ***9.2(9)*** *The storage capacity and usage of the potable water*

This COL item is addressed in Subsections 9.2.4.1, 9.2.4.2.2.1, 9.2.4.2.2.2 and 9.2.4.2.2.3.

CP COL 9.2(10) ***9.2(10)*** *State and Local Department of Health ~~of Natural Resources~~and Environmental Protection Standards* DCD_09.02.04-1

This COL item is addressed in Subsection 9.2.4.1.

CP COL 9.2(11) ***9.2(11)*** *Source of potable water to the site and the necessary required treatment*

This COL item is addressed in Subsections 9.2.4.1, 9.2.4.2.1 and Figure 9.2.4-1R.

CP COL 9.2(23) ***9.2(23)*** *The test and inspection requirements of the UHS*

This COL item is addressed in Subsection 9.2.5.4.

CP COL 9.2(24) ***9.2(24)*** *The required alarms, instrumentation and controls of the UHS system*

This COL item is addressed in Subsection 9.2.5.5.

CP COL 9.2(25) ***9.2(25)*** *The operating and maintenance procedures to address water hammer issues* DCD_09.02.01-12,13,14,30

This COL item is addressed in Subsections 9.2.1.2.1 and 13.5.2.1.

CP COL 9.2(26) ***9.2(26)*** *Maintenance and test procedures to monitor and flush out debris* DCD_09.02.01-17

This COL item is addressed in Subsections 9.2.1.2.1 and 13.5.2.1.

Chapter 10

Chapter 10 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_10.02.03-01	10.2	10.2-1	Response to RAI No. 6 Luminant Letter no.TXNB-09023 Date 06/17/2009	For FSAR Subsection 10.2.3.5, delete the entire paragraph and replace with the following: "A turbine maintenance and inspection procedure will be established prior to fuel load."	-
DCD_10.03.06-6	10.3.6.3.1	10.3-1	Reflect Response to DCD RAI No.250	Replace "industry guidelines" with "NSAC-202L-R3". Add new sentence to end of second paragraph.	3
RCOL2_10.03.06-1 RCOL2_10.03.06-2	10.3.6.3.1	10.3-1	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Replace "considers the information" with "addresses the concerns" and insert "consistent with the guidelines of" for the 2nd sentence of the 2nd paragraph. Replace the revision number for NSAC-202L from "R3" to "R2".	-
				Replace "a limited, but thorough, baseline inspection program" with "perform preservice inspection" for the first bullet in the 3rd paragraph.	-
	10.3.6.3.1.2	10.3-2	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Insert "to identify wall thickness margins for thinning and" in the 1st sentence of the 1st paragraph.	-
				Insert "with grid location" in the 2nd sentence of the 1st paragraph.	-
				Insert a new sentence after the 2nd sentence of the 1st paragraph.	-

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
RCOL2_10.03.06-1 RCOL2_10.03.06-2	10.3.6.3.1.2	10.3-2	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Delete the letter “s” in the word “inspections” and replace “are” with “after preservice inspection is” in the 3rd sentence of the 1st paragraph. Insert the word “trend” after “baseline” in the 3rd sentence of the 1st paragraph.	-
	10.3.6.3.1.4	10.3-3	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Insert a new bullet item after the 2nd bullet under “b. Implementing Procedures”.	-
				Insert “after plant operation cycles” at the end of the 4th bullet under “b. Implementing Procedures”.	-
	10.3.6.3.1.6	10.3-4	Response to RAI No. 7 Luminant Letter no.TXNB-09028 Date 8/7/2009	Insert new sentence after the 1st sentence.	-
RCOL2_10.03-1	10.3.2.3.2	10.3-1	Response to RAI No. 16 Luminant Letter no.TXNB-09033 Date 08/24/2009	Delete the entire Subsection 10.3.2.3.2 and its subsection subheading “Main Steam Safety Valves”.	-
RCOL2_10.03-1	10.3.7	10.3-4	Response to RAI No. 16 Luminant Letter no.TXNB-09033 Date 08/24/2009	Delete COL 10.3(2) description and state “Delete from DCD”.	-
RCOL2_10.04.08-2	10.4.8.2.1	10.4-6	Response to RAI No. 17 Luminant Letter no.TXNB-09034 Date 08/24/2009	Delete the entire second paragraph in FSAR Subsection 10.4.8.2.1.	-
DCD_10.03-1	10.3.2.4.3	10.3-1	Reflect Response to DCD RAI No. 329	Add new subsection.	4
DCD_10.03-1	10.3.7	10.3-4	Reflect Response	Add new COL item.	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			to DCD RAI No. 329		
DCD_10.04.07-1	10.4.7.7	10.4-5	Reflect Response to DCD RAI No. 124	Add new subsection.	4
DCD_10.04.07-1	10.4.12	10.4-9	Reflect Response to DCD RAI No. 124	Add new COL item.	4
HYDSV-16	10.4.5.3.2	10.4-5	Hydrology Site Safety Visit	Add new subsection.	4
HYDSV-16	10.4.5.6	10.4-5	Hydrology Site Safety Visit	Clarified the actuation of the makeup water pumps.	4

Chapter 11

Chapter 11 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00482	11.2.3.1	11.2-2	Editorial correction	Delete repeated phrase.	0
CTS-00481	Table11.2-14R	11.2-14	Editorial correction	Add “hr” in transit time.	0
MAP-11-001	11.3.3.3	11.3-2, 11.3-3	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 11.3(5)	0
CTS-00728	11.2.3.1	11.2-2	Clarification	Combined the statement of the second paragraph replacement and the statement of the last four paragraphs replacement.	4
CTS-00729	11.2.3.1	11.2-2	Editorial correction	Changed “to be” to” to remain”.	4
CTS-00805	11.2.3.1	11.2-2	Editorial correction	Separated the 5th paragraph. A new paragraph starts with the following sentence. “However, during the maximum…”.	4
HPSV-02	11.2.3.1	11.2-2 11.2-3	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Provided additional description about how discharge to Squaw Creek Reservoir will occur.	4
CTS-00730	11.2.3.1	11.2-3	Clarification	Added “CPNPP Units 3 and 4” in front of “waste holdup tanks” and “liquid effluent”.	4
HPSV-02	11.2.3.1	11.2-3	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Deleted commitment to evaluate circulating water dilution prior to Units 1 and 2 retirement.	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
HPSV-02	11.2.3.1	11.2-3	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Revised the description about the discharge line design.	4
CTS-00731	11.2.3.1	11.2-3	Editorial correction	Changed "…structure, system, and components…"to" …structures, systems, and components…"	4
CTS-00732	11.2.3.1	11.2-3	Editorial correction	Changed "…the local area rainfall and evaporation rate and half of liquid effluent." to" …the local area rainfall, evaporation rate, and receiving half of the CPNPP Units 3 and 4 liquid effluent."	4
CTS-00733	11.2.3.1	11.2-3	Editorial correction	Combined following sentences to one sentence to delete duplicate description. "The pond design includes a discharge line and transfer pump. A discharge line connects into CPNPP Units 1 and 2 circulating water return line to keep the pond from overflowing during periods of extreme weather conditions."	4
HPSV-02	11.2.3.4	11.2-4	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Added a new subsection to provide the evaporation pond design criteria and operating information.	4
HPSV-02	11.3.3.1	11.3-2	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Added note that noble gases are not present in evaporation pond.	4
HPSV-02	Figure 11.2-	11.2-25	NRC information need at HP Safety	Revised the figure to use dotted line for	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
	201(Sheet 9 of 9)		Site Visit (June 23 and 24,2009)	existing Unit 1 and 2 piping and a solid line for the evaporation pond.	
HPSV-04	11.3.3.1	11.3-2	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Corrected the discrepancy on total dose to skin and total body between the text and Table 11.3-9R.	4
HPSV-04	11.3.3.1	11.3-2	NRC information need at HP Safety Site Visit (June 23 and24,2009)	Identified maximum dose from the pond and the pond + the vent stack in text. Identified the h group organ pathway also.	4
HPSV-09	11.4.2.3	11.4-2	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Provided the additional description about the new low-level radwaste storage facility.	4
HPSV-10	11.5.2.9	11.5-2	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Revised to reflect that the ODCM will be re-written to apply to all four CPNPP units and to conform with the NEI.	4
CTS-00783	11.5.2.9	11.5-2	DCWG Meeting (July 16, 2009)	Deleted a following sentence. “CPNPP has already had an existing ODCM (Reference11.5-201) that is to reflect the new reactor units.”	5
CTS-00806	11.4.4.5	11.4-4	DCWG Meeting (July 16, 2009)	Added descriptions about mobile system connections and a commitment about the operational procedure.	5
CTS-00766	11.5.2.6	11.5-1	DCWG Meeting (July 16, 2009)	Add a following phrase between “These procedures” and “are prepared”. “, described in Subsection 13.5.2,”	5
CTS-00765	11.5.2.10	11.5-2	DCWG Meeting (July 16, 2009)	Deleted the following sentence. “CPNPP currently has a radiological	5

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
				environmental monitoring program for CPNPP Units 1 and 2 that is described in the plant Technical Specifications and the existing ODCM." Added the following sentence. "The radiological environmental monitoring program for CPNPP Units 3 and 4 follows the guidance of NEI 07-09."	

11.4.3.2 Process Control Program

CP COL 11.4(3) Replace the content of DCD Subsection 11.4.3.2 with the following.

This subsection adopts NEI 07-10, which is currently under review by the NRC staff. The Process Control Program (PCP) describes the administrative and operational controls used for the solidification of liquid or wet solid waste and the dewatering of wet solid waste. The purpose of the PCP is to provide the necessary controls such that the final disposal waste product meets applicable federal regulations (10 CFR Parts 20, 50, 61, 71, and 49 CFR Part 173), state regulations, and disposal site waste form requirements for burial at a low level waste disposal site that is licensed in accordance with 10 CFR Part 61. Waste processing (solidification and/or dewatering) equipment and services may be provided by third-party vendors. The process used in the existing design meets the applicable requirements of the PCP. Table 13.4-201 provides the milestone for PCP implementation.

Additional onsite radioactive solid waste storage is provided and is discussed in Subsection 11.4.2.3.

11.4.4.5 Mobile De-watering System

CP COL 11.4(4) Replace the last sentence in DCD Subsection 11.4.4.5 with the following.

CP COL 11.4(7)

The mobile de-watering station is vendor supplied and operated within the specific requirements and layout based on vendor specifications. The mobile system includes the necessary connections and fittings to the interface with the plant piping. The connectors are uniquely designed to prevent inadvertent cross connection between the radioactive and non-radioactive plant piping. The piping also includes backflow inhibitors. ~~Operating procedures will be developed and implemented with PCP so that the guidance and information in IE Bulletin 80-10 (Reference 11.4-29) is followed. The milestone for procedure implementation is listed in Table 13.4-201.~~Liquid effluent from the mobile de-watering station is routed to the Liquid Waste Management System and the non-condensables are vented to the A/B ventilation system. An operating procedure will be provided prior to fuel load to ensure proper operation of the mobile de-watering station to prevent the contamination of non-radioactive piping or uncontrolled releases of radioactivity into the environment. CTS-00806

11.5 PROCESS EFFLUENT RADIATION MONITORING AND SAMPLING SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

CP SUP 11.5(1) Add the following text to the end of the last paragraph in DCD Section 11.5.

Essential service water(ESW) pipe tunnel structure at elevation 793’-1’’ has been changed in site-specific layout. However, the location of process effluent radiation monitors in DCD Chapter 11 is not affected by the modification of ESW pipe tunnel structure, and Figures 11.5-2 can be used except for the structure of ESW pipe tunnel remains valid. The structure of ESW pipe tunnel is shown on Figure 1.2-2R.

11.5.2.6 Reliability and Quality Assurance

CP COL 11.5(4)
CP COL 11.5(5)

Replace the first sentence in the third paragraph in DCD Subsection 11.5.2.6 with the following.

The procedures for acquiring and evaluating samples of radioactive effluents, as well as procedures for inspection, calibration, and maintenance of the monitoring and sampling equipment are developed in accordance with RG 1.21 and RG 4.15. The procedures for the radioactive waste systems are developed in accordance with RG 1.33. The analytical procedures are developed in accordance with RG 1.21. These procedures, described in Subsection 13.5.2, are prepared and implemented under the quality assurance program referenced in Chapter 17. CTS-00766

11.5.2.7 Determination of Instrumentation Alarm Setpoints for Effluents

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CP COL 11.5(2) Replace the second sentence in DCD Subsection 11.5.2.7 with the following.

The methodology for the calculation of the alarm setpoints is part of the ODCM described in Subsection 11.5.2.9.

11.5.2.8 Compliance with Effluent Release Requirements

CP COL 11.5(4)
CP COL 11.5(5)

Replace the last sentence in DCD Subsection 11.5.2.8 with the following.

Site-specific procedures on equipment inspection, calibration, maintenance, and regulated record keeping, which meet the requirements of 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR 50 Appendix I, are prepared and implemented under the quality assurance program referenced in Chapter 17.

11.5.2.9 Offsite Dose Calculation Manual

Replace the first sentence in DCD Subection 11.5.2.9 with the following.

CP COL 11.5(2)
CP COL 11.5(1)

Fulfillment of the 10 CFR 50 Appendix I guidelines requires effluent monitor data. A description of the monitor controls and the calculation of the monitor setpoints are part of the ODCM. The ODCM also provides the rationale for compliance with the radiological effluent Technical Specifications and for the calculation of appropriate setpoints for effluent monitors. The ODCM follows the guidance of NEI 07-09. The ODCM and radiological effluent Technical Specifications, which reflect the new reactor units, are implemented in accordance with the milestone listed in Table 13.4-201. ~~CPNPP has already had an existing ODCM (Reference 11.5-201) that is to reflect the new reactor units.~~ The ODCM will be re-written to apply to all four CPNPP units and to conform with the NEI template before receipt of radioactive material in Unit 3 in accordance with FSAR Table 13.4-201.

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HPSV-10

11.5.2.10 Radiological Environmental Monitoring Program

CP COL 11.5(3)

Replace the content of DCD Subsection 11.5.2.10 with the following.

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~~CPNPP currently has a radiological environmental monitoring program for CPNPP Units 1 and 2 that is described in the plant Technical Specifications and the existing ODCM.~~ The program for CPNPP Units 3 and 4 is going to be described in the plant Technical Specification of CPNPP Units 3 and 4 and the ODCM, which reflect the new reactor units, is implemented in accordance with the milestone listed in Table 13.4-201. This program measures direct radiation using thermoluminescent dosimeters as well as analyses of samples of the air, water, vegetation, and fauna in the surrounding area. The guidance outlined in NUREG-1301 (Reference 11.5-21) and NUREG-0133 (Reference 11.5-18) is to be used when developing the radiological environmental monitoring program. The

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radiological environmental monitoring program for CPNPP Units 3 and 4 follows the guidance of NEI 07-09. CTS-00765

11.5.2.11 Site-Specific Cost-Benefit Analysis

CP COL 11.5(6) Replace the content of DCD Subsection 11.5.2.11 with the following.

The results of site-specific cost-benefit analysis are described in Subsections 11.2.1.5 and 11.3.1.5.

11.5.5 Combined License Information

Replace the content of DCD Subsection 11.5.5 with the following.

CP COL 11.5(1) ***11.5(1)*** *Site-specific aspects*

This COL item is addressed in Subsection 11.5.2.9.

CP COL 11.5(2) ***11.5(2)*** *Offsite dose calculation manual*

This COL item is addressed in Subsection 11.5.2.7 and 11.5.2.9.

CP COL 11.5(3) ***11.5(3)*** *Radiological and environmental monitoring program*

This COL item is addressed in Subsection 11.5.2.10.

CP COL 11.5(4) ***11.5(4)*** *Inspection, decontamination, and replacement*

This COL item is addressed in Subsections 11.5.2.6 and 11.5.2.8.

CP COL 11.5(5) ***11.5(5)*** *Analytical procedures*

This COL item is addressed in Subsections 11.5.2.6 and 11.5.2.8.

CP COL 11.5(6) ***11.5(6)*** *The site-specific cost benefit analysis*

This COL item is addressed in Subsection 11.5.2.11.

11.5.6 References

Add the following reference after the last reference in DCD Subsection 11.5.6.

Chapter 12

Chapter 12 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
DCD_12.01-2	12.1.3	12.1-2	Delete Outdated RG	Delete RG8.20, 8.26, and 8.32.	0
DCD_12.02-15	12.2.1.1.10	12.2-1	DCD_RAI 12.02-15	Add “40 CFR 190”.	0
CTS-00463	12.5	12.5-1	Clarification	Change description about entry into the interim waste storage building.	0
DCD_12.03-12.04-2	12.1.3	12.1-2	Reflect Response to DCD RAI No. 12.03-12.04-2	Add COL Items	3
CTS-00717	12.2.1.1.10	12.2-1	Clarification	Clarify description of Interim Radwaste Storage/Staging Building	4
HPSV-07	12.4.1.9.2.1	12.4-2	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Identified and added dose sources such as warehouse C, HIC yard.	4

Chapter 13

Chapter 13 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00484	13.1	13.1-17 13.1-18	Editorial correction	Change location of “Table 13.1-201 (Sheet 5 of 5)”.	0
CTS-00486	13.5	13.5-4 13.5-7	Editorial correction	Delete reference 13.5-201.	0
CTS-00488	13AA Table of Contents	13AA-ii	Editorial correction	Modify dot lines in Table of Contents.	0
CTS-00723	13.6	13.6-1	Reflect new rule	Add the new rule for the Cyber Security Plan.	4
CTS-00724	13.6	13.6-1	Update	Delete reference to NEI-03-12 for the physical security plan	4
CTS-00725	13.7	13.7-1	Update	Incorporate latest Rev.4 of NEI 06-06, “Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites”.	4
HPSV-09	13.2.1.1.3	13.2-1	NRC information need at HP Safety Site Visit (June 23 and 24,2009)	Added a subsection requiring initial and refresher Hazard Awareness Training.	4

Chapter 14

Chapter 14 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00635	14.2.2	14.2-1	Editorial correction	Change "Replace the last paragraph" to "Replace the last sentence of the second paragraph". Change "Appendix 14AA provides a description ...." to " A description .... are reconciled in Appendix 14AA".	0
RCOL2_14.03-1	14.2.12 14.2.12.1 14.2.13 Table 14.2-201	14.2-3 14.2-7 14.2-8	Responses to RAI No. 1 Luminant Letter TXNB-09010 Dated 5/1/2009	Add new item to ensure verification that local offsite fire departments utilize hose threads or adapters capable of connecting with onsite hydrants, hose couplings, and standpipe risers.	-
DCD_14.02-114	14.2.3 14.2.8.2.1 14.2.13	14.2-1 14.2-2 14.2-7	Reflect Response to DCD RAI No. 271.	Add description of STD COL 14.2(11) and STD COL 14.2(12) in accordance with DCD RAI No.271.	3
DCD_14.02-23	14.2.8.1	14.2-2	Reflect Response to DCD RAI No. 31.	Add description of STD COL 14.2(11) in accordance with DCD RAI No.31.	3
	14.2.13	14.2-7			
DCD_14.02-8	ACRONYMS AND ABBREVIATIONS	14-iv	Reflect Response to DCD RAI No.27	Add “Station Operations Review Committee”	4
	14.2.1	14.2-1	Reflect Response to DCD RAI No.27	Delete Subsection 14.2.1.	4
	14.2.2	14.2-1	Reflect Response to DCD RAI No.27	Delete reference to Appendix 14AA and revise text.	4
	14.2.3	14.2-1	Reflect Response to	Delete Subsection 14.2.3	4

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
			DCD RAI No.27		
	14.2.4 14.2.5 14.2.6	14.2-1 14.2-2	Reflect Response to DCD RAI No.27	Delete Subsection 14.2.4, 14.2.5 and 14.2.6.	4
	14.2.11	14.2-3	Reflect Response to DCD RAI No.27	Change COL information number	4
	14.2.13	14.2-7	Reflect Response to DCD RAI No.27	Revise COL information.	4
	Appendix 14AA		Reflect Response to DCD RAI No.27	Delete Appendix 14AA.	4
DCD_14.02-90	14.2.12	14.2-3	Reflect Response to DCD RAI No.93	Revise the description of replaced portion for COL information	4

Chapter 15

Chapter 15 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R

Chapter 16

Chapter 16 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R

Chapter 17

Chapter 17 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
CTS-00490	17.3	17.3-1	Editorial correction	Change description about quality assurance program.	0

Chapter 18

Chapter 18 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R

Chapter 19

Chapter 19 Tracking Report Revision List

Change ID No.	Section	FSAR Rev. 0 Page	Reason for change	Change Summary	Rev. of FSAR T/R
MAP-19-001	19.1.5.1.1	19.1-8 19.3-1	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 19.3(5)	0
MAP-19-002	19.2.5	19.2-1 19.3-1	Deletion of COL item. Letter MHI Ref:UAP-HF-08259, dated on Nov.7, 2008	Delete COL 19.3(6)	0
CTS-00491	ACRONYMS AND ABBREVIATIONS	19-v	Erratum	Change “Westuinghouse” to “Westinghouse”.	0
CTS-00714	19.2.5 19.2.7 19.3.3	19.2-1 19.2-4 19.3-1	Restoration of COL item. Letter MHI Ref: UAP-HF-09305 dated June10,2009	Restoration COL 19.3(6)	3

September 11 2009

Comanche Peak Nuclear Power Plant, Units 3 & 4
COL Application

Part 4

Technical Specifications

Update Tracking Report

Revision 0

Revision History

Revision	Date	Update Description
0	9/11/2009	Updated Chapters: Specifications, Bases

Introduction

Introduction – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R

Specifications

Specifications – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.0 # UAP-HF-09081 dated 03/06/2009 UTR Rev.1 # UAP-HF-09222 dated 04/30/2009 UTR Rev. 3 # UAP-HF-09413 dated 08/03/2009	0

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST - Analog (application of test for analog equipment)	~~For analog equipment, a~~An ACTUATION LOGIC TEST - Analog (the application of the test for analog equipment) shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST - Analog, as a minimum, shall include a continuity check of output devices.
ACTUATION LOGIC TEST (application of test for digital equipment, PSMS)	~~For the Protection and Safety Monitoring System (PSMS), a~~An ACTUATION LOGIC TEST is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation or the continuous self-test function. The PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. For the PSMS the self-testing is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.3 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.5. The software memory integrity test is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.4.1 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.4.1.c.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

1.1 Definitions

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. CHANNEL CALIBRATION encompasses devices that are subject to drift between surveillance intervals and all input devices that are not tested through continuous automated self-testing. Refer to TADOT for output devices that are not tested through continuous automated self-testing.

~~For analog measurements, the~~ CHANNEL CALIBRATION confirms the accuracy of the channel from sensor to digital Visual Display Unit (VDU) readout, as described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.4.2.

For analog measurements CHANNEL CALIBRATION confirms the analog measurement accuracy at five calibration setpoints corresponding to 0%, 25%, 50%, 75% and 100% of the instrument range. The confirmed setpoint are monitored on the safety VDUs.

For binary measurements, the CHANNEL CALIBRATION confirms the accuracy of the channel’s state change, as described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.4.~~2~~1.

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. A CHANNEL CHECK may be conducted manually or automatically. Where the CHANNEL CHECK is conducted automatically, an alarm shall be generated when the agreement criteria are not met.

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT) - Analog (application of test for analog equipment)

~~For analog equipment, a~~A COT - Analog shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT - Analog shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT - Analog may be performed by means of any series of sequential, overlapping, or total channel steps.

CHANNEL OPERATIONAL TEST (COT) (application of test for digital equipment, PSMS)

~~For the PSMS,~~A COT is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation~~, including digital Trip Setpoint values~~ or the continuous self-test function.

The PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses ~~all digital Trip Setpoints and trip functions. Digital Trip Setpoints are maintained in non-volatile software memory within each PSMS train~~data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. For the PSMS ~~T~~the self-testing is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.3 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.5. The software memory integrity test is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.4.1 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.4.1.c.

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit-specific document that provides cycle-specific parameter limits. These cycle-specific parameter limits shall be determined for each cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.

1.1 Definitions

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. ~~A TADOT does not include adjustment of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. Therefore, a TADOT is typically applicable only to binary devices that are not subject to drift. However, some binary devices that are subject to drift and are calibrated infrequently, may also require a TADOT, on a more frequent basis, to confirm gross operability.~~The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.
	There are two types of binary devices - those that have no drift potential, such as Manual Initiation switches and Actuation Output, and those that have drift potential, such as UV relays, valve position limit switches and RTB trip devices. The operability of binary devices that have drift potential is confirmed through CHANNEL CALIBRATION and/or RESPONSE TIME testing. The operability of devices that have no drift potential is confirmed through TADOT. The CHANNEL CALIBRATION confirms the accuracy of the device's binary state change with regard to its trip setpoint requirement (ie. the Allowable Value). The RESPONSE TIME test confirms the accuracy of the devices state change with regard to its trip timing requirement. The TADOT confirms only the state change operability (ie. there is no setpoint or timing accuracy confirmation needed). The TADOT includes adjustments as necessary, based on manufacture's recommendation, to maintain device reliability. For some binary devices with drift potential, a TADOT is specified in addition to the CHANNEL CALIBRATION and/or RESPONSE TIME test. The TADOT is specified on a more frequent basis than the CHANNEL CALIBRATION or RESPONSE TIME test, to confirm the state change operability of the devices, without checking its state change setpoint or timing accuracy. Checking the setpoint or timing accuracy more frequently than the CHANNEL CALIBRATION or RESPONSE TIME test interval is unnecessary, because the total channel uncertainty, including setpoint and/or timing drift between test intervals, is included in determination of the Nominal Setpoint, the Allowable Value and the response time requirement.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained
≥ 1.35 for typical hot cannel
≥ 1.33 for thimble hot channel
with WRB-2 DNB correlation and revised thermal design procedure (RTDP) .

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5072°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~2735~~2733.5 psig.

2.2 SAFETY LIMIT VIOLATIONS

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

2.2.2 If SL 2.1.2 is violated:

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

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

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, ~~and~~ LCO 3.0.8, and LCO 3.0.9.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours, b. MODE 4 within 13 hours, and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
LCO 3.0.4	When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

3.0 LCO Applicability

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient,"
LCO 3.1.4, "Rod Group Alignment Limits,"
LCO 3.1.5, "Shutdown Bank Insertion Limits,"
LCO 3.1.6, "Control Bank Insertion Limits," and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, ~~6~~ and 15.c, may be reduced to 3 required channels, provided:

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One required Manual Reactor Trip train inoperable.	B.1 Restore three trains to OPERABLE status.	72 hours
	OR	
	B.2 Be in MODE 3.	78 hours
C. One required Manual Reactor Trip ~~channel~~train inoperable.	C.1 Restore ~~three~~ train~~s~~ to OPERABLE status.	72 hours
	OR	
	C.2.1 Initiate action to fully insert all rods.	72 hours
	AND	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	73 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One High Power Range Neutron Flux (high setpoint) channel inoperable.	-------------------NOTE------------------- ~~The inoperable~~One channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment~~ of other channels~~. --	
	E.1.1 Place channel in trip.	72 hours
	AND	
	E.1.2 Reduce THERMAL POWER to ≤ 75% RTP.	78 hours
	OR	
	E.2.1 Place channel in trip.	72 hours
	AND	
	E.2.2 --------------NOTE-------------- Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ------------------------------------- Perform SR 3.2.4.2.	Once per 12 hours
	OR	
	E.3 Be in MODE 3.	78 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One required channel inoperable.	-------------------NOTE-------------------- For High Power Range Neutron Flux channels only, ~~the inoperable~~one channel may be bypassed for up to 12 hours for surveillance testing~~ of other channels~~. --		
	F.1	Place channel in trip.	72 hours
	OR		
	F.2	Be in MODE 3.	78 hours
G. One High Intermediate Range Neutron Flux channel inoperable.	G.1	Reduce THERMAL POWER to < P-6.	24 hours
	OR		
	G.2	Increase THERMAL POWER to > P-10.	24 hours
H. Two High Intermediate Range Neutron Flux channels inoperable.	H.1	--------------NOTE-------------- Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SDM. ------------------------------------	
		Suspend operations involving positive reactivity additions.	Immediately
	AND		
	H.2	Reduce THERMAL POWER to < P-6.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One required channel inoperable.	-------------------NOTE-------------------- Except for Pressurizer Pressure, Pressurizer Level, and SG Water Level, ~~the inoperable~~one channel may be bypassed for up to 12 hours for surveillance testing ~~of other channels~~. --	
	L.1 Place channel in trip.	72 hours
	OR	
	L.2 Reduce THERMAL POWER to < P-7.	78 hours
M. One required train inoperable.	--------------------NOTE------------------- One inoperable train may be bypassed for up to 4 hours for surveillance testing provided the other two trains are OPERABLE. --	
	M.1 Restore train to OPERABLE status.	24 hours
	OR	
	M.2 Be in MODE 3.	30 hours
N. One required RTB train inoperable.	N.1 Restore train to OPERABLE status.	24 hours
	OR	
	N.2 Apply the requirements of 5.5.18.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
S. Required Action and associated Completion Time for Condition N, Q, or R not met.	S.1 Be in MODE 3.	6 hours
T. Main Turbine Stop Valve Position channel inoperable	------------------NOTE------------------ One channel may be bypassed for up to 12 hours for surveillance testing. ---	
	T.1 Place channel in trip.	12 hours
	T.2 Reduce thermal power to < P-7	18 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	---------------------------NOTES------------------------- 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ~~2. Adjust nuclear instrument channel if absolute difference is > 1%.~~ ---	
	Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculations results exceed power range channel output by more than +2% RTP.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-----------------------------NOTE---------------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of the incore detector measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.4	Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1 (page 1 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip ~~i~~Initiation	1,2	3 trains	B	SR 3.3.1.4	NA	NA
	3[a], 4[a], 5[a]	3 trains	C	SR 3.3.1.4	NA	NA
2. High Power Range Neutron Flux						
a. high setpoint	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	±4% RTP	109% RTP
b. low setpoint	1[b],2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	±4% RTP	25% RTP
3. High Power Range Neutron Flux Rate						
a. Positive Rate	1,2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	±2% RTP	10% RTP with time constant ≥ 1 sec
b. Negative Rate	1,2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	±2% RTP[i]	7% RTP with time constant ≥ 1 sec
4. High Intermediate Range Neutron Flux	1[b], 2[c]	2	G,H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	±10% RTP[i]	25% RTP

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(i) An allowable value is not provided for time constants because time constants are digital values set in the application software. There is no drift or adjustments for these time constants.

NOTE: In all case, the values specified for Allowable Values and Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
7. Overpower ΔT	1,2	3	F	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.13	Refer to Note 2 after this table	Refer to Note 2 after this table
8. Pressurizer Pressure						
a. Low Pressurizer Pressure	1(e)	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±2.5% of span	1865 psig
b. High Pressurizer Pressure	1,2	3	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±2.5% of span	2385 psig
9. High Pressurizer Water Level	1(e)	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±3% of span	92% of span
10. Low Reactor Coolant Flow	1(e)	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9	±3% of rated flow	90% of rated flow
11. Low Reactor Coolant Pump (RCP) Speed	1(e)	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±0.5% rated pump speed	95.5% rated pump speed

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam Generator (SG) Water Level						
a. Low	1,2	3 per SG	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±3% of span	13% of span
b. High-High	$1^{(e)}$	3 per SG	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	±3% of span	70% of span
13. Turbine Trip						
a. Turbine Emergency Trip Oil Pressure	$1^{(e)}$	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 930 psig	1000 psig
b. Main Turbine Stop Valve Position	$1^{(e)}$	1 per valve	~~L~~T	SR 3.3.1.9 SR 3.3.1.12	≥ 1% open	5% open

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. ECCS actuation	1,2	3 trains	M	SR 3.3.1.5	NA	NA
15. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	O	SR 3.3.1.7 SR 3.3.1.10	±5% of span	1E-10 A
b. Low Power Reactor Trips Block, P-7	1	1 per train	P	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-10	1,2	4	O	SR 3.3.1.7 SR 3.3.1.10	±4% RTP	10%RTP
d. Turbine Inlet Pressure, P-13	1	~~4~~3	P	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9	±2.5% of span	10%~~RTP~~ Turbine Power
16. Reactor Trip Breakers (RTBs)	1,2	3 trains(i)	N,S	SR 3.3.1.4 SR 3.3.1.13	NA	NA
	3(b), 4(b), 5(b)	3 trains(i)	D	SR 3.3.1.4 SR 3.3.1.13	NA	NA

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(i) Two reactor trip breakers per train.

Table 3.3.1-1 (page 6 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
17. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	3 trains 1 each per RTB	Q,S	SR 3.3.1.4 SR 3.3.1.13	NA	NA
	$3^{(b)}$, $4^{(b)}$, $5^{(b)}$	3 trains 1 each per RTB	D	SR 3.3.1.4 SR 3.3.1.13	NA	NA
18. Automatic Trip Logic	1,2	3 trains	R,S	SR 3.3.1.5	NA	NA
	$3^{(b)}$, $4^{(b)}$, $5^{(b)}$	3 trains	D	SR 3.3.1.5	NA	NA

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function is initiated based on setpoints derived for DNB protection or core exit boiling conditions. Allowable Value shall not exceed the following nominal Trip Setpoints by more than [±5.6]% RTP (DNB Protection) or [±9.4]% RTP (Core Exit Boiling Limit).

$$\Delta T_{SP} = Lowselect(\Delta T_{SP1}, \Delta T_{SP2})$$

$$\Delta T \frac{(1 + T_7 s)}{(1 + T_8 s)} \left(\frac{1}{1 + T_9 s} \right) \geq \Delta T_{SP}$$

Where: T_7=[*]sec T_8=[*]sec T_9=[*]sec

1.DNB Protection

$$\cancel{\Delta T \frac{(1 + T_7 s)}{(1 + T_8 s)} \left(\frac{1}{1 + T_9 s} \right) \leq K_1 - K_2 \frac{(1 + T_2 s)}{(1 + T_3 s)} (T_{avg} - T_{avg0}) + K_3 (P - P_0) - f_1(\Delta I)}$$

$$\Delta T_{SP1} = \Delta T_0 \left(K_1 - K_2 \frac{(1 + T_2 s)}{(1 + T_3 s)} (T_{avg} - T_{avg0}) + K_3 (P - P_0) - f_1(\Delta I) \right)$$

Where: ΔT is measured RCS ΔT,°F.
ΔT_0 is indicated RCS ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T_{avg} is the measured RCS average temperature,°F.
T_{avg0} is the nominal T_{avg} at RTP, ≤ [*]°F.

P is the measured pressurizer pressure, psig
P_0 is the nominal RCS operating pressure, ≥ [*] psig

K_1 ≤ [*]	K_2 ≥ [*]/ F	K_3 ≥ [*]/psig
T_2 ≥ [*] sec	T_3 ≤ [*] sec	~~T_7 ≥ [*] sec~~
~~T_8 ≤ [*] sec~~	~~T_9 ≤ [*] sec~~	

$f_1(\Delta I)$ = [*] {[*] - (q_t - q_b)} when q_t - q_b ≤ [*]% RTP
0% of RTP when [*]% RTP < q_t - q_b ≤ [*]% RTP
[*] {(q_t - q_b) - [*]} when q_t - q_b > [*]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

*These values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT (continued)

2.Core Exit Boiling Limit

$$\Delta T\frac{(1+T_7 s)}{(1+T_8 s)}\left(\frac{1}{1+T_9 s}\right) \le K_4 - K_5\frac{(1+T_4 s)}{(1+T_5 s)}(T_{avg} - T_{avg0}) + K_6(P - P_0)$$

$$\Delta T_{SP2} = \Delta T_0\left(K_4 - K_5\frac{(1+T_4 s)}{(1+T_5 s)}(T_{avg} - T_{avg0}) + K_6(P - P_0)\right)$$

Where:ΔT is measured RCS ΔT,°F.

ΔT_0 is indicated RCS ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature,°F.

T' is the nominal T_{avg} at RTP, ≤ [*]°F.

P is the measured pressurizer pressure, psig

P_0 is the nominal RCS operating pressure, ≥ [*] psig

K_4 ≤ [*]	K_5 ≥ [*]/°F	K_6 ≥ [*]/psig
T_4 ≥ [*] sec	T_5 ≤ [*] sec	~~T_7 ≥ [*] sec~~
~~T_8 ≤ [*] sec~~	~~T_9 ≤ [*] sec~~	

*These values denoted with [*] are specified in the COLR.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required train inoperable.	--------------------NOTE------------------- One required train may be bypassed for up to 4 hours for surveillance testing provided the other required train(s) is OPERABLE. ---	
	C.1 Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 Be in MODE 5.	60 hours
D. One required channel inoperable.	------------------NOTE-------------------- ~~For Containment Pressure and Main Steam Line Pressure, the inoperable~~One channel may be bypassed for up to 12 hours for surveillance testing~~ of other channels~~. ---	
	D.1 Place channel in trip.	72 hours
	OR	
	D.2.1 Be in MODE 3.	78 hours
	AND	
	D.2.2 Be in MODE 4.	84 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. One required Containment High Area Radiation monitoring channel inoperable.	K.1 Restore channel to OPERABLE status. OR K.2.1 Be in MODE 3. AND K.2.2 Be in MODE 5.	72 hours 78 hours 108 hours
L. One or more ~~Automatic~~ Actuation Logic and Actuation Output trains inoperable. OR Two or more Containment High Range Area Radiation Monitoring channels inoperable. OR Required Action and associated Completion Time of Condition K not met.	L.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately
M. One or more Functions with one channel or train inoperable.	M.1 Place one train of the effected subsystem(s) ~~MCR Isolation train in emergency mode~~ in the emergency mode, depending on the inoperable train. --------------------NOTE-------------------- Inoperable train A or D affects both subsytem MCREFS and subsystem MCRATCS, while inoperable train B or C affects only subsystem MCRATCS. --	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One or more Functions with two channels or two trains inoperable.	N.1.1 Place ~~two MCR Isolation trains~~the effected subsystem(s) in the emergency mode.	Immediately
	AND	
	N.1.2 Enter applicable Conditions and Required Actions for ~~one CREFS train~~the effected subsystem(s) made inoperable by inoperable ~~CREFS~~ actuation instrumentation, depending on inoperable trains.	Immediately
	OR	
	N.2 Place all trains of the effected subsystem(s) in emergency mode.	Immediately
	--------------------Note-------------------- Inoperable train A or D affects both subsystem MCREFS and subsytem MCRATCS, while inoperable train B or C affects only subsystem MCRATCS. ---	
	~~----------------Note-------------------- This alternative is not available for failure of the Automatic Actuation Logic and Actuation Outputs. ---~~	
O. Required Action and associated Completion Time for Condition M or N not met in MODE 1, 2, 3, or 4.	O.1 Be in MODE 3.	6 hours
	AND	
	O.2 Be in MODE 5.	36 hours

Table 3.3.2-1 (page 1 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. ECCS Actuation						
a. Manual Initiation	1,2,3,4	3 trains	B	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3,4	3 trains	Q,R	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High Containment Pressure	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±2.8% of span	6.8 psig
d. Low Pressurizer Pressure	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±2.5% of span	1765 psig
e. Low Main Steam Line Pressure	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	525[(b)] psig

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

NOTE: In all case, the values specified for Allowable Values and Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

Table 3.3.2-1 (page 2 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2 switches per train for 3 trains	B	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3,4	3 trains	Q,R	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High-3 Containment Pressure	1,2,3	3	E	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±2.8% of span	34.0 psig

Table 3.3.2-1 (page 4 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Main Steam Line Isolation						
a. Manual Initiation	$1,2^{(h)},3^{(h)}$	Trains A and D	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	$1,2^{(h)},3^{(h)}$	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High-High Containment Pressure	$1, 2^{(h)}, 3^{(h)}$	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±2.8% of span	22.7 psig
d. Main Steam Line Pressure						
(1) Low Main Steam Line Pressure	$1, 2^{(h)}, 3^{(a)(h)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	$525^{(b)}$ psig
(2) High Main Steam Line Pressure Negative Rate	$3^{(f)(h)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	$100^{(g)}$ psi

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

(f) Below the P-11 (Pressurizer Pressure) interlock.

(g) Time constant utilized in the rate/lag controller is ≥ 50 seconds.

(h) Except when all MSIVs are closed.

Table 3.3.2-1 (page 5 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Main Feedwater Isolation						
5A. Main Feedwater ~~Control~~ Regulation valve Closure						
a. Low T_{avg}	1,2 (~~i~~j),3 (~~i~~j)	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±2°F	564°F
Coincident with Reactor Trip, P-4	Refer to Function 11.a for all P-4 requirements.					
5B. Main Feedwater Isolation						
a. Manual Initiation	1,2 (i),3 (i)	Trains A and D	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1, 2 (i), 3(i)	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High-High SG Water Level	1,2 (i),3(a) (i)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	70% of span
d. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					

(a) Above the P-11 (Pressurizer Pressure) interlock.

(i) Except when all MFIVs, MFRVs, MFBRVs, and SGWFCVs are closed.

(j) Except when all MFRVs are closed.

Table 3.3.2-1 (page 6 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Emergency Feedwater Actuation						
a. Manual Initiation	1,2,3	3 trains	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3	3 trains	J,T	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. Low SG Water Level	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	13% of span
d. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					
e. LOOP Signal	1,2,3	3 per bus for each EFW train	F	SR 3.3.2.5 SR 3.3.2.7 SR 3.3.2.8	~~±1.5% of span~~ ≥ 4830 V with a time delay of ≤ 0.8 second	~~4727~~4934 V[(l)] with ≤ 2 sec time delay
f. Trip of all Main Feedwater Pumps	1,2	1 per pump	H	SR 3.3.2.6 SR 3.3.2.8	NA	NA

(l) Nominal Trip Setpoint

Table 3.3.2-1 (page 7 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
7. Emergency Feedwater Isolation						
a. Manual Initiation	1,2,3	2 trains per SG	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3	2 trains per SG	G	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High SG Water Level	1,2,3 (a)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	50% of span
~~coincident~~ Coincident with Reactor Trip, P-4	Refer to Function 11.a for all P-4 requirements.					
and ~~no~~No Low Main Steam Line Pressure	Refer to Function 7.d for all initiation functions and requirements.					
d. Low Main Steam Line Pressure	1,2,3 (a)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	525 psig
8. CVCS Isolation						
a. Manual Initiation	1,2,3	Trains A and D	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3	Trains A and D	G	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. High Pressurizer Water Level	1,2,3 (a)	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	92% of span

(a) Above the P-11 (Pressurizer Pressure) interlock.

Table 3.3.2-1 (page 8 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
9. Turbine Trip						
a. Actuation Logic and Actuation Outputs	1,2,3	4 trains	G	SR 3.3.2.2 SR 3.3.2.4	NA	NA
b. Reactor Trip, P-4	Refer to Function 11.a for all P-4 requirements.					
c. High-High SG Water Level ~~(P-14)~~	1,2 (i),3 (i)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	70% of span
10. Reactor Coolant Pump Trip						
a. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					
Coincident with Reactor Trip, P-4	Refer to Function 11.a for all P-4 requirements.					
11. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	3 trains	F	SR 3.3.2.9	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	I	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7	±2.5% of span	1915 psig

(i) Except when all MFIVs, MFRVs, MFBRVs, and SGWFCVs are closed.

Table 3.3.2-1 (page 9 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
12. Containment Purge Isolation						
a. Containment Isolation Phase A - Manual Initiation	Refer to Function 3.a. (Containment Isolation Phase A - Manual Initiation) for all initiation functions and requirements.					
b. Containment Spray - Manual Initiation	Refer to Function 2.a. (Containment Spray - Manual Initiation) for all initiation functions and requirements.					
c. Actuation Logic and Actuation Outputs	1,2,3,4	Trains A and D	L	SR 3.3.2.2 SR 3.3.2.4	NA	NA
d. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					
e. ~~Containment Radiation~~ Containment High Range Area Radiation	1,2,3,4	3	K, L	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±6% of span	100 R/h
13. Main Control Room (MCR) Isolation						
a. Manual Initiation	1,2,3,4,(~~a~~k)	~~4 trains~~3 trains including A and D[m]	M, N, O, P	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Output	1,2,3,4,(~~a~~k)	~~4 trains~~3 trains including A and D[m]	M, N, O,P	SR 3.3.2.2 SR3.3.2.4	NA	NA
c. MCR Outside Air Intake Radiation						
(1) MCR Outside Air Intake Gas Radiation	1,2,3,4,(~~a~~k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±6% of span	2E-6 μCi/cc
(2) MCR Outside Air Intake Particulate Radiation	1,2,3,4,(~~a~~k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±6% of span	8E-10 μCi/cc
(3) MCR Outside Air Intake Iodine Radiation	1,2,3,4,(~~a~~k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±6% of span	8E-10 μCi/cc

Table 3.3.2-1 (page 9 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
d. ECCS Actuation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.					

(~~a~~k)During movement of irradiated fuel assemblies within containment.

(m) Two trains of MCREFS are required to be operable (trains A and D); three trains of MCRATS are required to be operable (three out of four trains A, B, C, D).

Table 3.3.2-1 (page 10 of 10)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. Block Turbine Bypass and Cooldown Valves						
a. Manual Initiation	1,2 (i), 3 (i)	Trains A and D	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2 (i), 3 (i)	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. Low-low T_{avg} Signal	1,2 (i), 3 (i)	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	2.0°F	553°F

(i) Except when all MSIVs are closed.

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTES--

~~1.LCO 3.0.4 not applicable~~

~~2~~1.Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	---------------NOTE--------------- 1. For RCS Hot and Cold Leg Temperatures, this Condition is applicable only if at least one channel (Hot or Cold) is operable in each loop. Otherwise, go to Condition C. 2. For SG Water Level and EFW flow, this condition is applicable only if at least one channel (Level or flow) is operable in each loop. Otherwise, go to Condition C. ------------------------------------- Initiate action in accordance with Specification 5.6.5.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with two required channels inoperable.	C.1	---------------NOTE--------------- 1. For RCS Hot and Cold Leg Temperatures, this Condition is applicable only when both channels in the same loop are inoperable. 2. For SG Water Level and EFW flow, this condition is applicable only when both channels in the same loop are inoperable. --------------------------------------	
		Restore one channel to OPERABLE status.	7 days
	OR		
	C.2	--------------NOTE------------- This alternate action may be used ~~O~~only ~~required to be performed~~ when the Emergency Feedwater Pit Level is inoperable. ------------------------------------	
		Apply the requirements of Specification 5.5.18.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1	Enter the Condition referenced in Table 3.3.3-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3-1.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	12 hours
F. As require by Required Action D.1 and referenced in Table 3.3.3-1.	F.1	Initiate action in accordance with Specification 5.6.5.	Immediately

Table 3.3.3-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.Wide Range Neutron Flux	2	E
2.Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	1 per loop [d]	~~E~~F
3.RCS Cold Leg Temperature (Wide Range)	1 per loop[d]	~~E~~F
4.RCS Pressure (Wide Range)	2	E
5.Reactor Vessel Water Level	2	~~E~~F
6.Containment Pressure	2	E
7.Containment Isolation Valve Position	2 per penetration flow path[a][b]	E
8.Containment High Range Area Radiation	2	~~E~~F
9.Pressurizer Water Level	2	E
10.Steam Generator Water Level (Wide Range)	1 per steam generator[d]	~~E~~F
11.Steam Generator Water Level (Narrow Range)	2 per steam generator	E
12.Core Exit Temperature - Quadrant 1	2[c]	E
13.Core Exit Temperature - Quadrant 2	2[c]	E
14.Core Exit Temperature - Quadrant 3	2[c]	E
15.Core Exit Temperature - Quadrant 4	2[c]	E
16.Emergency Feedwater Flow	1 per SG[d]	~~E~~F
17.Degrees of Subcooling	2	E
18.Main Steam Line Pressure	2 per steam generator	E
19.Emergency Feedwater Pit Level	2	E
20.Refueling Water Storage Pit Level (Wide Range)	2	E
21.Refueling Water Storage Pit Level (Narrow Range)	2	E

(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 channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two core exit thermocouples.

(d) A RCS hot leg temperature wide range and a RCS cold leg temperature wide range of the same train are pair PAM function. A SG water level wide range and an emergency feedwater flow of the same train are pair PAM function. The idea is to treat parameters forming a pair as one set and choose the number of required channels to be two, providing a basis for control.

3.3 INSTRUMENTATION

3.3.5 Loss of Power (LOP) Class 1E Gas Turbine Generator (GTG) Start Instrumentation

LCO 3.3.5 Three channels per required bus of the loss of voltage Function and three channels per required bus of the degraded voltage Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated Class 1E GTG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one or more channels per required bus inoperable.	A.1	--------------NOTE-------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ------------------------------------ Place channel in trip.	6 hours
B. One or more Functions with two or more channels per required bus inoperable.	B.1	Restore all but one channel per bus to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Enter applicable Condition(s) and Required Action(s) for the associated Class 1E GTG made inoperable by LOP Class 1E GTG start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2	Perform TADOT for LOP undervoltage relays.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.3	Perform CHANNEL CALIBRATION for LOP undervoltage relays with Nominal Trip Setpoint and Allowable Value as follows: a. Loss of voltage Allowable Value ≥ 4830 V with a time delay of ≤ 2 second Loss of voltage Nominal Trip Setpoint ~~4727~~4934 V with a time delay of 2 second. b. Degraded voltage Allowable Value ≥ 6210 V with a time delay of ≤ 10 seconds. Degraded voltage Nominal Trip Setpoint 6314 V with a time delay of 10 seconds.	In accordance with the Surveillance Frequency Control Program

NOTE: In all case, the values specified for Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

Table 3.3.6-1 (page 1 of 2)
Diverse Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Reactor Trip/ Turbine Trip/ MFW Isolation						
a. Manual Initiation	1,2,3[a]	1[b]	A	SR 3.3.6.5 SR 3.3.6.6	NA	NA
b. Automatic Actuation Logic and Actuation Outputs	1,2,3[a]	2	A	SR 3.3.6.4 SR 3.3.6.5	NA	NA
c. Low Pressurizer Pressure	1,2,3[a]	2	A	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.3	~~±2.5% of span~~≥ 1805 psig	1825 psig
d. High Pressurizer Pressure	1,2,3[a]	2	A	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.3	~~±2.5% of span~~≤ 2445 psig	2425 psig
e. Low Steam Generator Water Level	1,2,3[a]	1 per SG for any 2 SGs	A	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.3	~~±3% of span~~≥ 4% span	7% of span
f. Rod Drive Motor-Generator ~~set~~Set	1,2,3[a]	2(1 for each MG-~~set~~ Set)	A	SR 3.3.6.6	NA	NA
2. EFWS Actuation						
a. Manual Initiation	1,2,3[a]	1[b]	A	SR 3.3.6.5	NA	NA
b. Automatic Actuation Logic and Actuation Outputs	1,2,3[a]	2	A	SR 3.3.6.5	NA	NA
c. Low Steam Generator Water Level	Refer to Function 1.e for all Low Steam Generator Water Level requirements.					

(a) With the Pressurizer Pressure > P-11

(b) Manual initiation functions require operation of 2 switches, the Permissive Switch for DAS HSI and the manual initiation switch on the DHP.

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~~Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation~~.

--NOTES--

1. All reactor coolant pumps (RCPs) and CS/RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

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

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

2. No RCP shall be started with any RCS cold leg temperature ≤ the Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND	Immediately

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required RHR or RCS loops ~~is~~are in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 13% for required RCS loops.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 Two residual heat removal (CS/RHR) loops shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE or

b. The secondary side water level of at least two steam generators (SGs) shall be ≥ 13%.

--NOTES--

1. The CS/RHR pumps of the loops in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

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

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

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loops ~~is~~are OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ the Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR unless the secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures.

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS Loops Filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RHR loop inoperable. <u>OR</u> One or more required SGs with secondary side water level not within limit <u>AND</u> Two RHR loops OPERABLE and in Operation.	A.1 Initiate action to restore a third RHR loop to OPERABLE status <u>OR</u> A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
B. Less than two RHR loops OPERABLE or in operation. .	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. <u>AND</u> B.2 Initiate action to restore two RHR loops to OPERABLE status and operation.	Immediately Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Three residual heat removal (RHR) loops shall be OPERABLE and two RHR loops shall be in operation.

---NOTES---

1. One CS/RHR pump may be removed from operation for ≤ 15 minutes when switching from one loop to another provided:

   a. The core outlet temperature is maintained > 10°F below saturation temperature,

   b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other two RHR loops are OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12

An LTOP System shall be OPERABLE with a maximum of two Safety Injection (SI) pumps and one charging pump capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities:

a. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 456 psig and ≤ 484 psig, or

b. The RCS depressurized and an RCS vent of ≥ ~~2.6~~4.7 square inches.

---NOTES---

1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swap operations.

2. Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY:

MODE 4 when any RCS cold leg temperature is ≤ LTOP arming temperature specified in the PTLR,

MODE 5,

MODE 6 when the reactor vessel head is on.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One required RHR suction relief valve inoperable in MODE 4, 5, 6.	E.1 Restore required RHR suction relief valve to OPERABLE status.	12 hours
	OR	
	E.2 Depressurize RCS and establish RCS vent of ≥ ~~2.6~~4.7 square inches	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of two SI pumps are capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.3	Verify each accumulator is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	Verify RHR suction valve is open for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.5	-------------------------------NOTE-------------------------- Only required to be performed when complying with LCO 3.4.12.b -- Verify required RCS vent ≥ ~~2.6~~4.7 square inches open.	Unlocked open vent valve(s) AND other vent path(s) OR in accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. ~~1~~0.5 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. RHR suction valve interlock function inoperable	C.1	Isolate the affected penetration by use of one closed manual or deactivated automatic valve	4 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level) monitor,

b. One containment atmosphere radioactivity monitor (~~gaseous or~~ particulate), and

c. One containment air cooler condensate flow rate monitor.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>	
	A.2 Restore required containment sump monitor to OPERABLE status.	30 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits.~~RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 ~~≥ 1.0 µCi/gm~~not within limit.	--------------------NOTE-------------------- LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 ~~≥ 300 µCi/gm~~not within limit.	--------------------NOTE-------------------- LCO 3.0.4.c is applicable --	~~6~~48 hours
	B.1 ~~Be in MODE 3 with T_{avg} < 500°F~~Restore DOSE EQUIVALENT XE-133 to within limit.	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1 AND C.2	Be in MODE 3 ~~with T_{avg} < 500°F~~. Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	------------------------------NOTE------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 300 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	------------------------------NOTE------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ ~~19,300~~19,338 gallons and ≤ ~~19,700~~19,734 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 586psig and ≤ 695 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 4000 ppm and ≤ 4200 ppm.	In accordance with the Surveillance Frequency Control Program AND --------NOTE-------- Only required to be performed for affected accumulators ------------------------ Once within 6 hours after each solution volume increase of ≥ 190 gallons ~~of indicated level~~ that is not the result of addition from the refueling water storage pit
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ 1920 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position (with power to the valve operator removed).		In accordance with the Surveillance Frequency Control Program
Number	Function	Position	
~~2~~SIS-AOV-201B and C	Accumulator Makeup	CLOSED	
SR 3.5.2.2	Verify each SIS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.		In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify each SI pump's developed head at the test flow point is greater than or equal to the required developed head.		In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each SI pump starts automatically on an actual or simulated actuation signal.		In accordance with the Surveillance Frequency Control Program
SR 3.5.2.5	Verify by visual inspection, each SIS train ECC/CS STRAINER is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.		In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3Each containment isolation valve shall be OPERABLE.

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

ACTIONS

---NOTES---

1.Penetration flow path(s), except for 36 inch high volume purge valve 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 containment isolation valves. -------------------------------- 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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTE------------ Only applicable to penetration flow paths with two containment isolation valves. -------------------------------- One or more penetration flow paths with two containment isolation valves inoperable for reasons other than Condition D.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ------------NOTE------------ Only applicable to penetration flow paths with only one containment isolation valve and a closed system. -------------------------------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	OR		
	C.1.2	-------------NOTES------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18.	72 hours
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more penetration flow paths with one or more high volume purge valves not within purge valve leakage limits.	D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	AND	
	D.2 -------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND	
	D.3 Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 36 inch high volume purge valve is sealed closed, except for one high volume purge valve in a penetration flow path while in Condition D of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 8 inch low volume purge valve is closed, except when the 8 inch containment low volume 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.	In accordance with the Surveillance Frequency Control Program
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.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	------------------------------NOTE------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for 36 inch high volume purge valves with resilient seals.	184 days AND Within 92 days after opening the valve
SR 3.6.3.~~6~~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.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Three containment spray (CS) trains shall be OPERABLE.

---NOTES---

CS train may be considered OPERABLE during alignment and operation for decay heat removal as RHRS if capable of being manually realigned to the CS mode of operation.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required containment spray train inoperable.	A.1 Restore three containment spray trains to OPERABLE status. <u>OR</u> A.2 ------------NOTES------------ This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18	~~7 days~~<u>72 hours</u> ~~7 days~~<u>72 hours</u>
~~B. One or less required containment spray trains OPERABLE.~~	~~B.1 Restore two containment spray trains to OPERABLE status.~~	~~72 hours~~
~~C~~<u>B</u>.Required Action and associated Completion Time of Condition A or B not met.	~~C~~<u>B</u>.1 Be in MODE 3. <u>AND</u> ~~C~~<u>B</u>.2 Be in MODE 5.	6 hours 36 hours

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Six MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~One or more MSSVs per steam generator inoperable~~One or more steam generators with one or more MSSVs inoperable.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND	4 hours

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

VALVE NUMBER				LIFT SETTING (psig ± ~~3~~1%)
STEAM GENERATOR				
#1	#2	#3	#4	
MS-V509A	MS-V509B	MS-V509C	MS-V509D	1185
MS-V510A	MS-V510B	MS-V510C	MS-V510D	1215
MS-V511A	MS-V511B	MS-V511C	MS-V511D	1244
MS-V512A	MS-V512B	MS-V512C	MS-V512D	1244
MS-V513A	MS-V513B	MS-V513C	MS-V513D	1244
MS-V514A	MS-V514B	MS-V514C	MS-V514D	1244

3.7 PLANT SYSTEMS

3.7.4 Main Steam Depressurization Valves (MSDVs)

LCO 3.7.4 Four MSDV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MSDV line inoperable.	A.1 Restore required MSDV line to OPERABLE status.	7 days
	OR	
	A.2 Apply the requirements of Specification 5.5.18.	7 days
B. Two or more required MSDV lines inoperable.	B.1 Restore all but one MSDV line to OPERABLE status.	24 hours
	OR	
	B.2 Apply the requirements of Specification 5.5.18.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	~~24~~12 hours

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater System (EFWS)

LCO 3.7.5 Four EFW trains shall be OPERABLE with all EFW pump discharge cross-connect line isolation valves in all trains closed.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to one turbine driven EFW pump inoperable. <u>OR</u> ------------NOTE------------ Only applicable if MODE 2 has not been entered following refueling. -------------------------------- One turbine driven EFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status. <u>OR</u> --------------NOTE-------------- When the EFW pump discharge cross-connect line isolation valves are closed. ------------------------------------- A.2 Open all EFW pump discharge cross-connect line isolation valves.	7 days 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One ~~required~~ EFW train inoperable in MODE 1, 2, or 3 for reasons other than Condition A.	B.1	Restore EFW train to OPERABLE status. OR --------------NOTE-------------- When the EFW pump discharge cross-connect line isolation valves are closed. -------------------------------------	72 hours
	B.2	Open all EFW pump discharge cross-connect line isolation valves.	72 hours
C. Required Action and associated Completion Time for Condition A or B not met. OR Two ~~required~~ EFW trains inoperable in MODE 1, 2, or 3.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	~~18~~12 hours
D. Three EFW trains inoperable in MODE 1, 2, or 3.	D.1	--------------NOTE-------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one additional EFW train is restored to OPERABLE status. ------------------------------------- Initiate action to restore one additional EFW train to OPERABLE status.	Immediately

3.7 PLANT SYSTEMS

3.7.6 Emergency Feedwater Pit (EFW Pit)

LCO 3.7.6 Two EFW Pits shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both EFW Pits inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2.1 Restore both EFW Pits to OPERABLE status.	7 days
	OR	
	A.2.2 Apply the requirements of Specification 5.5.18.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	~~24~~12 hours

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water System (ESWS)

LCO 3.7.8 Three ESWS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESWS train inoperable.	A.1 --------------NOTES------------- ~~1. Enter applicable and Required Actions of LCO 3.8.1, "AC Sources - Operating," for Class 1E gas turbine generator made inoperable by ESWS.~~ 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ESWS. ------------------------------------- Restore three ESWS trains to OPERABLE status.	72 hours
	OR	
	A.2 -------------NOTES------------- This Required Action is not applicable in MODE 4. ------------------------------------- Apply the requirements of Specification 5.5.18.	72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	-------------------------------NOTE-------------------------- Isolation of ESWS flow to individual components does not render the ESWS inoperable. --- Verify each ESWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each ESWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. The motor operated valve provided at the discharge of each pump opens automatically after starting the ESW pump. This interlock prevents the pump from starting if the valve is not closed. The closed discharge valve opens after starting the ESWP.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each ESWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required MCRVS inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	C.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	C.2	Verify mitigating actions to ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	C.3	Restore CRE boundary to OPERABLE status.	90 days
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours
E. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	E.1	Place OPERABLE MCRVS trains in emergency mode.	Immediately
	OR		
	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
F. Required MCRVS inoperable during movement of irradiated fuel assemblies. OR Required MCRVS inoperable due to inoperable CRE boundary during movement of irradiated fuel assemblies.	F.1	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each MCREFS train for ≥ 10 continuous hours with the heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required MCREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each MCRVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program
SR 3.7.10.5	Verify two MCRATCS trains have the capacity to remove the ~~assumed~~design heat load.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.12 Fuel Storage Pit Water Level

LCO 3.7.12 The fuel storage pit water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the fuel storage pit water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	At the start of any spent fuel movement campaign AND In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.15 Main Steam Line Leakage

LCO 3.7.15 Main steam line leakage through the pipe walls inside containment shall be limited to 0.5 gpm.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Main steam line leakage exceeds Operational limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify main steam line leakage into the containment Sump ≤ 0.5 gpm.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following ac electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E ac electrical power distribution system,

b. Three Class 1E Gas Turbine Generators (GTGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and

c. The associated automatic load sequencers for each required Class 1E GTG shall be OPERABLE.

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

ACTIONS

--NOTE--
LCO 3.0.4.b is not applicable to Class 1E GTGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One required offsite circuit inoperable.~~	~~A.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit.~~	~~1 hour~~ ~~AND~~ ~~Once per 8 hours thereafter~~
	~~AND~~	
	~~A.2.1 Restore required offsite circuit to OPERABLE status.~~	~~72 hours~~
	~~OR~~	
	A.2.2 -------------NOTE-------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable AND C.2.1 Restore one required offsite circuit to OPERABLE status. OR C.2.2 -------------NOTE-------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features 24 hours 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours
H. Two offsite circuits and one or more required GTGs inoperable. OR One offsite circuit and two or more required GTGs inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	Verify each Class 1E GTG starts from standby condition and achieves: a. In ≤ 100 seconds, voltage ≥ ~~6210~~6762 V and frequency ≥ 59.4 Hz and b. Steady state voltage ≥ 6762 V and ≤ 7038 V, and frequency ≥ 59.4 Hz and ≤ 60.6 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------------------NOTE--------------------------- This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify each Class 1E GTG's noncritical automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal. a. Overspeed b. Generator differential current, and c. High exhaust gas temperature	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more required Class 1E GTG inoperable.	B.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	B.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.4 Initiate action to restore required Class 1E GTGs to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	------------------------------NOTE--------------------------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.8 through SR 3.8.1.10, SR 3.8.1.12 through SR 3.8.1.15, and SR 3.8.1.17. --- For ac sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.7, SR 3.8.1.11, SR 3.8.1.16, SR 3.8.1.18, and SR 3.8.1.19, are applicable.	In accordance with applicable SRs

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
E.	One or more Class 1E GTGs with starting air receiver pressure < ~~270~~398 psig and ≥ ~~185~~228 psig.	E.1	Restore starting air receiver pressure to ≥ 270 psig.	48 hours
F.	Required Action and associated Completion Time not met. OR One or more Class 1E GTGs with gas turbine fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1	Declare associated Class 1E GTG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 91,000 gallons of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 81 gallons.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the GTG Fuel Oil Testing Program.	In accordance with the GTG Fuel Oil Testing Program
SR 3.8.3.4	Verify each Class 1E GT/G air start receiver pressure is ≥ ~~270~~398 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 Inverters in three trains shall be OPERABLE.

---NOTE---

One inverter may be disconnected from its associated dc bus for ≤ 24 hours to perform an equalizing charge on its associated battery, provided:

a. The associated ac vital bus is energized from its Class 1E transformer, and

b. All other ac vital buses are energized from their associated OPERABLE inverters.

Train A and B or Train C and D ac vital buses shall not be supplied from Class 1E transformer concurrently.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 --------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any ac vital bus de-energized. ------------------------------------- Restore inverter to OPERABLE status.	24 hours
	OR	
	A.2 --------------NOTE------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18	24 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 The ac, dc, and ac vital bus electrical power distribution subsystems shall be OPERABLE as specified in Table 3.8.9-1.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more~~ required ac electrical power distribution subsystems inoperable.	--------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for dc trains made inoperable by inoperable power distribution subsystems. --	
	A.1 Restore required ac electrical power distribution subsystem(s) to OPERABLE status.	8 hours
	OR	
	A.2 --------------NOTE------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18.	8 hours
B. One ~~or more~~ required ac vital buses inoperable.	B.1 Restore ac vital bus subsystem(s) to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	One ~~or more~~ required dc electrical power distribution subsystems inoperable.	C.1	Restore required dc electrical power distribution subsystem(s) to OPERABLE status.	2 hours
D.	Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
		AND		
		D.2	Be in MODE 5.	36 hours
E.	Two or more required electrical power distribution subsystems inoperable that result in a loss of safety function.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required ac, dc, and ac vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portions of ac, dc, and ac vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

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

ACTIONS

--NOTE--

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required ac, dc, or ac vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One or more required ac, dc, or ac vital bus electrical power distribution subsystems inoperable.~~	~~A.1 Declare associated supported required feature(s) inoperable.~~	~~Immediately~~
	~~OR~~	
	~~A.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~A.2.2 Suspend movement of irradiated fuel assemblies.~~	~~Immediately~~
	~~AND~~	
	~~A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.~~	~~Immediately~~
	~~AND~~	
	A.2.4 Initiate actions to restore required ac, dc, and ac vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.4 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	A.5 Close one door in each air lock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify two RHR loops are in operation and circulating reactor coolant at a flow rate of ≥ ~~2400~~2645 gpm per pump.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.5.1 Close each penetrations providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent. <u>OR</u> B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify two RHR loops are in operation and circulating reactor coolant at a flow rate of ≥ ~~2400~~2645 gpm per pump.	In accordance with the Surveillance Frequency Control Program
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.8 Decay Time

LCO 3.9.8 The reactor shall be subcritical for ≥ 24 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 24 hours.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify that the reactor has been subcritical for ≥ 24 hours by verification of the date and time of subcriticality.	Start movement of irradiated fuel assemblies within containment.

4.0 DESIGN FEATURES

4.1 Site Location

The site area is approximately 7,700 acres located in Somervell County in North Central Texas. Squaw Creek Reservoir extends into Hood County. The site is situated along Squaw Creek, a tributary of the Paluxy River, which is a tributary of the Brazos River. The site is over 30 miles southwest of the nearest point in Fort Worth and approximately 4.5 miles north-northwest of Glen Rose, the nearest community.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 257 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with ZIRLO cladding, which is a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium based alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Rod Cluster Control Assemblies

The reactor core shall contain 69 Rod Cluster Control Assemblies (RCCAs) each with 24 ~~rodlets/RCCA~~rods per assembly. The RCCA adsorber material shall be silver indium cadmium as approved by the NRC.

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~FSAR Chapter 9~~Subsection 9.1.1 of the FSAR,

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 200 ppm which includes an allowance for uncertainties as described in Subsection 9.1.1 of the FSAR, and

d. A nominal 11.1 inch center to center distance between fuel assemblies placed in spent fuel storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~FSAR Chapter 9~~Subsection 9.1.1 of the FSAR,

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~FSAR Chapter 9~~Subsection 9.1.1 of the FSAR, and

d. A nominal 16.9 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pit below 23 ft above the top of irradiated fuel assemblies seated in the storage racks.

4.3.3 Capacity

The spent fuel storage pit is designed and shall be maintained with a storage capacity limited to no more than 900 fuel assemblies.

5.2 ORGANIZATION

5.2.2 Unit Staff (continued)

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician and chemistry technician shall be on site when fuel is in the reactor. The positions may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required positions.

~~d) Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), radiation protection technicians, plant equipment operators, and key maintenance personnel).~~

~~The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.~~

~~Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.~~

~~Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.~~

d. The Shift Operations Manager shall hold an SRO license.

e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, and AG-1.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Main Control Room Emergency Filtration System (MCREFS)	3600 cfm
Annulus Emergency Exhaust System (AEES)	5600 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
MCREFS	3600 cfm

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

ESF Ventilation System	Penetration	RH	~~Face Velocity~~
MCREFS	2.5%	70%	~~2400 fps~~

5.5.16 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by approved exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

   2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 57.5 psig. The containment design pressure is 68 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criteria are:

      a. Overall air lock leakage rate is≤ 0.05 L_a when tested at ≥ P_a.

      b. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.18 Configuration Risk Management Program (CRMP) (continued)

c. This program shall satisfy all the requirements ~~have the following as~~ specified in NEI 06-09 including, but not limited to, the following:

1. Station procedure of the CRMP process with specifying the station functional organizations and personnel responsible for each action of CRMP implementation,

2. Training of responsible personnel,

3. PRA model to meet the technical adequacy requirement of NEI 06-09,

4. Appropriate CRM tool.

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Bases

Bases – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.0 # UAP-HF-09081 dated 03/06/2009 UTR Rev.1 # UAP-HF-09222 dated 04/30/2009 UTR Rev. 3 # UAP-HF-09413 dated 08/03/2009	0

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

The above criterion a. is represented by the limit DNBR values stated in SL 2.1.1.1, which are determined using WRB-2 DNB correlation with RTDP and include uncertainties of DNB correlation and key input parameters relevant to DNBR analysis as described in Reference 2.

The Reactor Trip System setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in Ref. 4) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

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

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

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

BASES

SAFETY LIMITS (continued)

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

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

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

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

REFERENCES

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

2. ~~FSAR Chapter~~FSAR Subsection 4.4.1.1

3. ~~FSAR Chapter 7~~FSAR Subsection 7.2.1.

4. FSAR Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

a. Main steam relief valve,

b. Turbine Bypass,

c. Rod Control System,

d. Pressurizer Water Level Control, or

e. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is ~~2735~~2733.5 psig.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.~~8~~9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

LCO 3.0.6 (continued)

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1),

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2), or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.6 (continued)

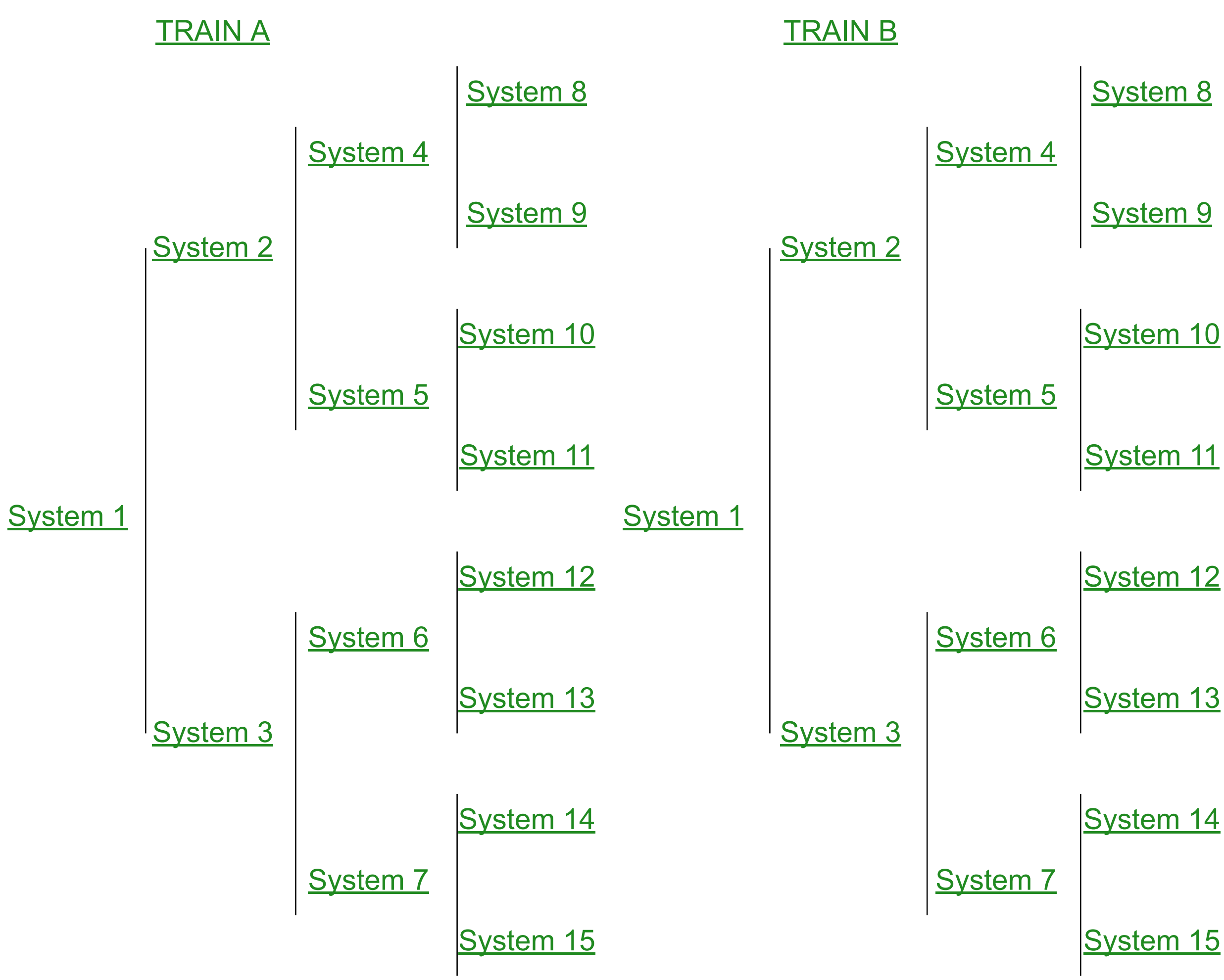

Figure B 3.0-1
Configuration of Trains and Systems

LCO 3.0.6 (continued)

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABITLITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

BASES

LCO 3.0.8 (continued)

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

BASES

LCO 3.0.9 (continued)

---------------------------------------Reviewer's Note---------------------------------------
LCO 3.0.9 may be expanded to other initiating event categories provided plant-specific analysis demonstrates that the frequency of the additional initiating events is bounded by the generic analysis or if plant-specific approval is obtained from the NRC.

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the

BASES

LCO 3.0.9 (continued)

affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

SR 3.0.1 (continued)

to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. Safety injection (SIS) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with SIS considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

APPLICABLE SAFETY ANALYSES (continued)

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $K_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration,

b. Control bank position,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration, and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR ~~Chapter 15~~Section 15.1 and 15.4.

3. 10 CFR 100.

BASES

BACKGROUND (continued)

critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to

BASES

APPLICABLE SAFETY ANALYSES (continued)

the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is least negative near BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR ~~Chapter 15~~Subsection 15.0.2.5 and 15.4.3.

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. LCOs 3.1.5 and 3.1.6 ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2, and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

BASES

ACTIONS (continued)

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

3. FSAR ~~Chapter 15~~Section 15.1, 15.4 and Subsection 15.0.0.2.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two or more groups to provide for precise reactivity control. A ~~group~~group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two or more groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits and overlap patterns are specified in the COLR. An example is provided for information only in Figure B 3.1.6-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.6-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 135 steps for a fully withdrawn position of 265 steps. The fully withdrawn position is defined in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii), in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.

2. 10 CFR 50.46.

3. FSAR ~~Chapter 15~~Section 15.1, 15.4 and Subsection 15.0.0.2.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indications to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms (CRDMs). The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two or more groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) which is included in the CRDM control system and the Rod Position Indication (RPI) System.

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the RPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indication channels satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii). The control rod position indications monitor control rod position, which is an initial condition of the accident.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR ~~Chapter 15~~Section 15.1, 15.4 and Subsection 15.0.0.2.3.

BACKGROUND (continued)

operation, and after each refueling. The PHYSICS TESTS requirements for reloaded fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Power Distribution – Intermediate Power,

b. Power Distribution – Full Power, and

c. ~~Critical Boron Concentration – Full Power~~HZP to HFP reactivity difference.

These tests are performed in MODE 1. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. The last two tests are performed at ≥ 90% RTP.

a. The Power Distribution – Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels at least one time by 30% RTP and between 40% and 80% RTP. This test uses the incore flux detectors to measure core power distribution. The requirements for the Flux Symmetry Test described in ANSI/ANS-19.6.1-2005 (Ref. 4) are satisfied by the Power Distribution Test.

BASES

BACKGROUND (continued)

b. The Power Distribution – Full Power Test measures the power distribution of the reactor core at ≥ 90% RTP using incore flux detectors.

c. The ~~Critical Boron Concentration – Full Power Test~~HZP to HFP reactivity difference simply measures the critical boron concentration at > 90% RTP, with all rods fully withdrawn, the lead control bank being at or near its fully withdrawn position, and with the core at equilibrium xenon conditions.

For initial startups, there are two currently required tests that violate the referenced LCO. The Axial Flux Difference Instrumentation Calibration Test and Axial Power Distribution Oscillation Test, performed at approximately 50% and 75% RTP, require large axial flux difference that exceed the limits specified in the relevant LCO. And the Rod Cluster Control Assembly Misalignment Measurement and Radial Power Distribution Oscillation Test, performed at approximately 50% RTP, require individual rod misalignments that exceed the limits specified in the relevant LCO.

APPLICABLE SAFETY ANALYSES

The fuel is protected by an LCO, which preserves the initial conditions of the core assumed during the safety analyses. The methods for development of the LCO, which are superseded by this LCO, are described in Ref. 5. The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating controls or process variables to deviate from their LCO limitations.

FSAR ~~Chapter 14~~ Section 14.2(Ref. 6) defines requirements for initial testing of the facility, including PHYSICS TESTS. The zero, low power, and power tests are summarized in this ~~chapter~~section. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-2005 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limits,” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” or LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR)” are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, “Heat Flux Hot Channel Factor ($F_Q(Z)$),” and LCO 3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),” are satisfied, power level is maintained ≤ 85% RTP, and SDM is within the limits specified in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, LCO 3.1.8 requires surveillance of the hot channel factors and SDM to verify that their limits are not being exceeded.

PHYSICS TESTS include measurements of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(~~d~~c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1, to verify certain core physics parameters. The power level is limited to ≤ 85% RTP and the power range neutron flux trip setpoint is set at 10% RTP above the PHYSICS TESTS power level with a maximum setting of 90% RTP. Violation of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.3, or LCO 3.2.4, during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the requirements of LCO 3.2.1 and LCO 3.2.2 are satisfied and provided:

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 3, March, 2007.
4. ANSI/ANS-19.6.1-2005, November 29, 2005
5. MUAP-07026-P, “Mitsubishi Reload Evaluation Methodology”, December, 2007
6. FSAR ~~Chapter 14~~Section 14.2.

BASES

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn,

~~b) Critical Boron Concentration - Control Rods Inserted,~~

b. Control Rod Worth, and

c. Isothermal Temperature Coefficient (ITC)

These tests are performed in MODE 2. These and other supplementary tests may be required to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

~~b) The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% Δk/k when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron~~

BASES

BACKGROUND (continued)

~~reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."~~

b. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

c. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the

BASES

BACKGROUND (continued)

temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in Ref. 5. The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

FSAR ~~Chapter 14~~Section 14.2 (Ref.6) defines requirements for initial testing of the facility, including PHYSICS TESTS. The zero, low power, and power tests are summarized in this ~~chapter~~section. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-2005 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP the reactor coolant temperature is kept ≥ 541°F, and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(~~d~~c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One power range neutron flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, ~~6~~ and 15.c may be reduced to 3 required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration,

g. Isothermal temperature coefficient (ITC), when below the zero power testing range,

h. Moderate defect, when above the zero power testing range, and

i. Doppler defect, when above the zero power testing range.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the zero power testing range, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 3, March, 2007.

4. ANSI/ANS-19.6.1-2005, November 29, 2005.

5. MUAP-07026-P, “Mitsubishi Reload Evaluation Methodology”, December, 2007

6. FSAR ~~Chapter 14~~Section 14.2.

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 230 cal/gm (Ref. 2), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$F_Q(Z) \leq (CFQ / P)$ for $P > 0.5$

$F_Q(Z) \leq (CFQ / 0.5)$ for $P \leq 0.5$

where:CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR, and

P = THERMAL POWER / RTP

BASES

SURVEILLANCE REQUIREMENTS (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.46, 1974.

2. FSAR ~~Chapter 15~~Subsection 15.0.0.1.2

3. 10 CFR 50, Appendix A, GDC 26.

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification,"

6. February 1994.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of is $F^N_{\Delta H}$ allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being

REFERENCES	1.	FSAR ~~Chapter 15~~Subsection 15.0.0.1.2.
	2.	10 CFR 50, Appendix A, GDC 26.
	3.	10 CFR 50.46.

BASES

BACKGROUND (continued)

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^{N}_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Refs. 1 and 2) entails:

a. Establishing an envelope of allowed power shapes and power densities,

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes,

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics, and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in the safety analyses. This ensures that fuel cladding integrity is maintained for the postulated accidents. AOOs, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS (continued)

Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.

SR 3.2.3.3

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. FSAR ~~Chapter 15~~Subsection 15.0.0.2.3.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

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

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

REFERENCES

1. 10 CFR 50.46.

2. FSAR ~~Chapter 15~~Subsection 15.0.0.1.2.

3. 10 CFR 50, Appendix A, GDC 26.

BACKGROUND (continued)

Technical Specifications contain ~~channel~~measured accuracy values related to the OPERABILITY of equipment required for safe operation of the facility. The ~~channel~~measured accuracy value accommodates expected drift in the analog components of the channel that would have been specifically accounted for in the setpoint methodology for calculating the Trip Setpoint 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 recalibrate the device 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 value needs to be specified in the Technical Specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value. The Allowable Value is another important component of the LSSS.

The Allowable Value specified in Table 3.3.1-1 serves as the LSSS such that a channel is OPERABLE if the ~~channel~~measured accuracy is found not to exceed the Allowable Value during CHANNEL CALIBRATION. The CHANNEL CALIBRATION verifies the instrument at five calibration setpoints corresponding to 0%, 25%, 50%, 75% and 100% of the instrument range. As such, the Allowable Value accounts for the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is "OPERABLE" under these circumstances, the channel ~~accuracy~~ should be left adjusted to a value within the established channel calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual accuracy of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a technical specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

BASES

BACKGROUND (continued)

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB),

2. Fuel centerline melt shall not occur, and

3. The RCS pressure SL of ~~2735~~2733.5 psig shall not be exceeded. |

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into ~~four~~three distinct but interconnected modules as illustrated in FSAR Chapter 7 (Ref. 2), and as identified below: |

1. Field transmitters, process sensors or field contacts: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured,

2. The RPS, including Nuclear Instrumentation System (NIS): provides signal conditioning, analog to digital conversion, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to the reactor trip breakers (RTBs), and digital output to control board/control room/miscellaneous VDUs, and

Reactor trip breakers (RTBs): provide the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor.

BASES

BACKGROUND (continued)

Generally, if a parameter is used for input to the protection circuits and a control function, three channels with a two-out-of-three logic are also sufficient to provide the required reliability and redundancy. The Signal Selection Algorithm (SSA) within the PCMS ensures the control systems can withstand an input failure to the control system without causing erroneous control system operation which would otherwise require the protection function actuation. Since the input failure does not cause an erroneous control system action that challenges the protection function, the input failure is considered a single failure in the RTS and the RTS remains capable of providing its protective function with the remaining two operable channels. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-603-1991 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

The RTB trains are arranged in a two out of four configuration. Therefore, three logic trains are required to ensure no single random failure of a logic train will disable the RTS. The logic trains are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions allow removing logic trains from service during maintenance.

Allowable Values and RTS Setpoints

The Trip Setpoints used in the digital bistables are based on the Analytical Limits defined in the accident analysis and the channel uncertainty. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints specified in Table 3.3.1-1 in the accompanying LCO are conservative with respect to the Analytical Limits. The methodology used to calculate the Allowable Values and Trip Setpoints incorporates all of the known uncertainties applicable to each channel (Ref. 2). The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint and corresponding Allowable Value. The Trip Setpoint entered into the digital bistable is more conservative than that specified by the Analytical Limit (LSSS) to account for measurement errors detectable by the CHANNEL CALIBRATION. The Allowable Value serves as the Technical Specification OPERABILITY limit for the purpose of the CHANNEL CALIBRATION. One example of such a change in measurement error is drift during the surveillance interval. If the measured accuracy does not exceed the Allowable Value, the channel is considered OPERABLE.

BASES

BACKGROUND (continued)

The Trip Setpoint is the value at which the bistable is set. The Trip Setpoint value ensures the LSSS and the safety analysis limits are met for surveillance interval selected when a channel is adjusted based on the stated channel uncertainties. Any channel is considered to be properly adjusted when the "as left" accuracy value is within the band for CHANNEL CALIBRATION uncertainty allowance (i.e. ± instrument, signal conditioning, and A/D converter uncertainties). The Trip Setpoint value (i.e. expressed as a value without inequalities) is used for the purposes of COT.

Trip Setpoints consistent with the requirements of the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of PAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or PA and the equipment functions as designed).

Digital Trip Setpoints are maintained in non-volatile software memory within each RPS train. Each train of the process control equipment is self-tested continuously on line to verify that the digital Trip Setpoint settings are correct. Trip Setpoints are also verified periodically through a diverse software memory integrity test, which may be conducted with the RTS train out of service. A designated instrument channel is taken out of service for periodic calibration. SRs for the channels and trains are specified in the SRs section.

NOTE: The Allowable Value in Table 3.3.1-1 is the maximum deviation at the calibration setpoints that can be measured during CHANNEL CALIBRATION. This value is included in the calculations that determined the TRIP SETPOINT in Table 3.3.1-1. The “expected as-found value” shall be as specified in the plant-specific setpoint analysis. The expected as-found value reflects the expected normal drift of actual plant equipment, so that a degraded device can be identified before the Allowable Value limit is reached.

Reactor Trip Breakers

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. There are eight RTBs, two from each of four RTB trains, arranged in a two out of four configuration.

During normal operation the output from the RPS is a voltage signal that energizes the undervoltage coils in the RTBs. When protective action is required, the RPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-

BASES

BACKGROUND (continued)

energized undervoltage coil, and the RTBs are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the RPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 2. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has built in self-testing that automatically tests the decision logic Functions while the unit is at power. When any one or two trains are taken out of service for testing, the other two trains are capable of providing unit monitoring and protection until the testing has been completed.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of PAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 3 and 9 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. A channel is OPERABLE provided the "as-found" ~~accuracy~~ value, measured during surveillance testing, does not exceed its associated Allowable Value. For digital functions Allowable Values are defined in terms pertinent to the channel calibration setpoints. For analog functions Allowable Values are defined in terms pertinent to the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO generally requires OPERABILITY of three or two channels in each instrumentation Function, three trains of Manual Reactor Trip in each logic Function, and three trains in each Automatic Trip Logic Function. Three OPERABLE instrumentation channels in a two-out-of-three configuration are required when one RTS channel is also used as a control system input. The SSA within the control system prevents the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. The input failure is considered a single failure in the RTS and RTS remains capable of providing its protective function with the remaining two operable channels.~~In this case, the RTS will still provide protection, even with random failure of one of the other two protection channels.~~ The SSA ensures there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three configuration allows one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

In all cases where the LCO states “Restore channel or train to OPERABLE status”, this means restore the required number of channels or trains to OPERABLE status. Therefore, restoration of an alternate channel or train, other than the failed channel or train, is also acceptable.

In Table 3.3.1-1, the values specified for Allowable Values and Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip initiation

   The Manual Reactor Trip initiation ensures that the control room operator can initiate a reactor trip at any time by using any two out of four hardwired reactor trip switches in the control room. A Manual Reactor Trip initiation accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The High Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events.

In MODE 2 when below the P-6 setpoint and in MODES 3, 4, and 5 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident, the High Source Range Neutron Flux trip must be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. Above the P-6 setpoint, the High Intermediate Range Neutron Flux trip and the High Power Range Neutron Flux (low setpoint) trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the High Source Range Neutron Flux trip may be manually bypassed which will also de-energize the NIS source range detectors ~~are de-energized~~. Above the P-10 setpoint, the High Source Range Neutron Flux trip is automatically bypassed and the NIS source range detectors are automatically de-energized.

In MODES 3, 4, and 5 with all rods fully inserted and the Rod Control System not capable of rod withdrawal, and in MODE 6, the outputs of the Function to RTS logic are not required OPERABLE.

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is initiated based on setpoints derived for DNB protection or core exit conditions. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include all pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit or core exit boiling is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The reactor trip occurs if indicated loop ΔT exceeds the lower setpoint of the DNB protection limit setpoint and the core exit boiling limit setpoint. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Low Reactor Coolant Flow

The Low Reactor Coolant Flow trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-7 setpoint, the reactor trip on low flow in ~~two or more~~any one RCS loop~~s~~ is automatically enabled. Each RCS loop has four flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Low Reactor Coolant Flow channels per loop to be OPERABLE in MODE 1 above P-7.

In MODE 1 above the P-7 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

11. Low Reactor Coolant Pump (RCP) Speed

The Low RCP Speed trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The speed of each RCP is monitored. Above the P-7 setpoint a low speed detected on two or more RCPs will initiate a reactor trip. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires three Low RCP Speed channels (one channel per loop) to be OPERABLE in MODE 1 above P-7. One channel per loop is sufficient for this trip Function because the Low RCS Flow trip alone provides sufficient protection of unit SLs for loss of flow events. The Low RCP Speed trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no power distributions are expected to occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more loops is automatically enabled.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. High-High SG Water Level

The High-High SG Water Level trip Function ensures that protection is provided against an excessive cooldown due to increase in feedwater flow. An increase in the feedwater flow rate will cause an increase in SG water level and reduction in the reactor coolant temperature. Reduction in the coolant temperature adds reactivity as a result of the positive moderator density coefficient, thereby increasing the reactor power.

This Function also performs the ESFAS functions of generating ~~a P-14 interlock signal, initiating~~ a turbine trip~~,~~ and initiating main feedwater isolation.

The LCO requires three channels of the High-High SG Water Level trip ~~and the High-High trip~~ Function to be OPERABLE in MODE 1 above P-7. The trip Function is automatically enabled on increasing power by the P-7 interlock and automatically blocked on decreasing power once the P-7 interlock is cleared. Although the High-High SG Water Level trip is blocked ~~B~~below the P-7 setpoint, ~~this condition will result in~~the ESFAS functions ~~actuations~~ to trip the turbine and isolate Main Feedwater are OPERABLE above and below the P-7 setpoint. These ESFAS functions allow~~ing~~ the operator sufficient time to evaluate plant conditions and take corrective actions.

13. Turbine Trip

a. Turbine Emergency Trip Oil Pressure

The Turbine Emergency Trip Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-7 setpoint, approximately ~~50~~10% power, will not actuate a reactor trip. Four pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-four pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the High

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Pressurizer Pressure trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Emergency Trip Oil Pressure to be OPERABLE in MODE 1 above P-7.

Below the P-7 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Emergency Trip Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip - Main Turbine Stop Valve Position

The Main Turbine Stop Valve Position trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level ~~below~~above the P-7 setpoint. This action will ~~not~~ actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will normally operate in the presence of a single failure due to redundant limit switches on each valve. However this trip Function is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the High Pressurizer Pressure trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Emergency Trip Oil Pressure turbine trip Function. Each main turbine stop valve is equipped with two limit switches that input to the RTS. If the limit switches indicate that all four stop valves are closed, a reactor trip is initiated.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Main Turbine Stop Valve Position channels, one per valve, to be OPERABLE in MODE 1 above P-7. One channel on each valve must trip to cause reactor trip.

Below the P-7 setpoint, a load rejection can be accommodated by the Turbine Bypass System. In MODE 2,

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the High Power Range Neutron Flux (low setpoint) and High Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

d. Turbine Inlet Pressure, P-13

The Turbine Inlet Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the ~~rated full~~turbine power ~~pressure~~. This is determined by two-out-of-four pressure detectors. The LCO requirement for this Function ensures that three of the inputs to the P-7 interlock are available.

The LCO requires three channels of Turbine Inlet Pressure, P-13 interlock to be OPERABLE in MODE 1.

The Turbine Inlet Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

16. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires three OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Rod Control System. Thus, the train consists of two main breakers. Three OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

17. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 19 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

18. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 16 and 17) and Automatic Trip Logic (Function 18) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs ~~and associated bypass breakers~~ to open and shut down the reactor.

The LCO requires three trains of RTS Automatic Trip Logic to be OPERABLE. Having three OPERABLE trains ensures that random failure of a single logic train will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 8).

BASES

ACTIONS (continued)

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 72 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (78 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems.

With the unit in MODE 3, ACTION C would apply to any inoperable Manual Reactor Trip Function if the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

The initial completion time of 72 hours is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19. The manual reactor trip function remains fully operable from the Safety VDUs, even when one Manual Reactor Trip channel is inoperable.

C.1, C.2.1, and C.2.2

Condition C applies to the Manual reactor trip Function in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

- Manual Reactor Trip

This action addresses the train orientation for ~~these~~one Function~~s~~. With one train inoperable, the inoperable train must be restored to OPERABLE status within 72 hours. If the affected Function~~(s)~~ cannot be restored to OPERABLE status within the allowed 72 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the same 72 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, ~~these~~this Function~~s are~~is no longer required.

The Completion Time of 72 hours is reasonable considering that there are three automatic actuation trains and two other Manual Reactor Trip trains OPERABLE, and the low probability of an event occurring during this interval. The completion time also considers that the manual reactor trip function, for the inoperable Manual Reactor Trip train, can also be actuated

ACTIONS (continued)

from the Safety VDU for that train. Therefore, the ability to initiate a manual reactor trip through safety related equipment remains functional in all three required trains.

The initial completion time of 72 hours is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19. The manual reactor trip function remains fully operable from the Safety VDUs, even when one Manual Reactor Trip channel is inoperable.

D.1, D.2.1, and D.2.2

Condition D applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted:

- RTBs,
- RTB Undervoltage and Shunt Trip Mechanisms, and
- Automatic Trip Logic.

This action addresses the train orientation for these Functions. With one train inoperable, the inoperable train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the two remaining OPERABLE trains are adequate to perform the safety function, and given the low probability of an event occurring during this interval. The Completion Time also considers that the two remaining OPERABLE trains each have continuous self-testing and redundant RTBs.

E.1.1, E.1.2, E.2.1, E.2.2, and E.3

Condition E applies to the Power Range Neutron Flux (high setpoint) Function.

BASES

ACTIONS (continued)

The NIS power range detectors provide input to the Rod Control System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is justified because the remaining three OPERABLE channels have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS.

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 78 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 72 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels < 75% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight hours are allowed to place the plant in MODE 3. The 78 hour Completion Time includes 72 hours for channel corrective maintenance, and an additional 6 hours for the MODE reduction as required by Required Action ~~D~~E.3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

One channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment. The 12 hours bypass limit is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

BASES

ACTIONS (continued)

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 12 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified based on operating experience.

Required Action E.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

F.1 and F.2

Condition F applies to the following reactor trip Functions:

- Power Range Neutron Flux (low setpoint),
- Overtemperature ΔT,
- Overpower ΔT,
- Power Range Neutron Flux Positive Rate,
- Power Range Neutron Flux Negative Rate,
- High Pressurizer Pressure, and
- Low SG Water Level.

BASES

ACTIONS (continued)

A known required inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic (for the trip functions where the required number of operable channels is three) or one-out-of three logic (for the tri functions where the required number of operable channels is four) for actuation of the two-out-of-~~four~~N trips, where N is three or four (depending on the required number of operable channels). The 72 hours allowed to place the inoperable channel in the tripped condition is justified because the remaining two operable channels (for the trip functions where the required number of operable channels is three) or the remaining three operable channels (for the trip functions where the required number of operable channels is four) have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The number of Required Channels for the High Power Range Neutron Flux Rate is four. Four channels are required because each channel measures neutron flux in one quadrant of the core. Anomalies occurring in one core quadrant can be seen by the neutron flux detector in that quadrant and by the neutron detectors in the two adjacent quadrants, but not by the deterctor in the opposite quadrant. So to ensure event detection and accomodate a single failure, neutron flux detectors must be operable in all four quadrants.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour is based on operating experience.

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

One channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment. The 12 hour bypass limit is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR

ACTIONS (continued)

Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

G.1 and G.2

Condition G applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

H.1 and H.2

Condition H applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

ACTIONS (continued)

Required Action H.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Required Action ~~H~~I.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

K.1, K.2.1, and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour.

BASES

ACTIONS (continued)

L.1 and L.2

Condition L applies to the following reactor trip Functions:

- Low Pressurizer Pressure,
- High Pressurizer Water Level,
- Low Reactor Coolant Flow,
- Low Reactor Coolant Pump Speed,
- High-High SG Water Level,
- Turbine Trip – Turbine Emergency Trip Oil Pressure, and
- Turbine Trip – Main Turbine Stop Valve Position.

With one required channel inoperable, the inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition when above the P-7 setpoint results in a partial trip condition requiring only one additional channel to initiate a reactor trip.

These Functions do not have to be OPERABLE below the P-7 setpoint because there is insufficient heat production to generate DNB conditions below the P-7 setpoint. The 72 hours allowed to place the channel in the tripped condition is justified because the remaining two operable channels have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channels, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition L.

Except for Pressurizer Pressure, Pressurizer Level, and SG Water Level, one channel may be bypassed for up to 12 hours for surveillance testing. The 12 hours bypass limit is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6a.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

BASES

ACTIONS (continued)

This bypass is not allowed for Pressurizer Pressure, Pressurizer Level, and SG Water Level because these channels are also used for control. If a failure were to occur in one of the two remaining control channels, a plant transient could occur that would require a plant trip, but a plant trip would not occur with only one remaining operable channel.

The initial completion time of 72 hours is justified in the PSMS reliabilty analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

M.1 and M.2

Condition M applies to the ECCS actuation input ~~and the RTS Automatic Trip Logic~~ in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one required train inoperable, 24 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 24 hours is reasonable considering that in this Condition, the two remaining OPERABLE trains are adequate to perform the safety function and given the low probability of an event during this interval. The 24 hours allowed to restore the train to OPERABLE status also considers that the two remaining OPERABLE trains each have automatic self-testing as described for ACTUATION LOGIC TEST. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one inoperable train up to 4 hours for surveillance testing, provided the other two trains are OPERABLE.

N.1 and N.2

Condition N applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one required train inoperable, 24 hours is allowed for train corrective maintenance to restore the train to OPERABLE status. The 24 hour Completion Time is reasonable considering that in this Condition, the two remaining OPERABLE trains are adequate to perform the safety function and given the low probability of an event during this interval. Required Action N.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

The initial completion time of 24 hours is justified in the PSMS reliability

ACTIONS (continued)

analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

O.1 and O.2

Condition O applies to the P-6 and P-10 interlocks. With one or more required channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

P.1 and P.2

Condition P applies to the P-7 and P-13 interlocks in MODE 1. With one or more required channels/trains inoperable for two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

Q.1 and Q.2

Condition Q applies to the RTB Undervoltage and Shunt Trip Mechanisms, i.e. diverse trip features, in MODES 1 and 2. For one RTB in a required train with one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours.

The Completion Time of 48 hours for Required Action Q.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, one OPERABLE RTB in the affected RTB train and two OPERABLE RTB trains capable of performing the safety function and given the low probability of an event occurring during this interval. Required

BASES

ACTIONS (continued)

Action Q.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

R.1 and R.2

Condition R applies to ~~the ECCS actuation input and~~ the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one required train inoperable, 24 hours are allowed to restore the train to OPERABLE status. The Completion Time of 24 hours is reasonable considering that in this Condition, the two remaining OPERABLE trains are adequate to perform the safety function and given the low probability of an event during this interval. The 24 hours allowed to restore the train to OPERABLE status also considers that the two remaining OPERABLE trains each have automatic self-testing as described for ACTUATION LOGIC TEST. Required Action R.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

The Required Actions have been modified by a Note that allows bypassing one inoperable train up to 4 hours for surveillance testing, provided the other two trains are OPERABLE.

S.1

Condition S applies when the Required Action and associated Completion Time for Condition N, Q, or ~~S~~R have not been met. If the train cannot be returned to OPERABLE status, the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

Placing the unit in MODE 3 from Condition N results in Condition ~~C~~D entry while an RTB is inoperable.

(From Condition Q) With the unit in MODE 3, ~~ACTION~~Condition D would apply to any inoperable RTB trip mechanism.

T.1 and T.2

Condition T applies to Main Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition requiring three additional channels to initiate a reactor trip. If the channel can not be restored to OPERABLE status or placed in the trip condition, then power must be reduced below P-7 setpoint within the next 6 hours. The 6 hours allowed for reducing power is consistent with other power

BASES

ACTIONS (continued)

reduction action completion times.

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. These times are justified because this is an anticipatory trip that is note credited in the safety analysis, and a diverse turbine trip also initiated from the Turbine Emergency Oil Pressure.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies all trains of the RTS. However, when testing a Channel, it is only necessary to manually verify that the channel is OPERABLE in its respective train. This is because the interface to other trains is continuously verified through self-testing. Self-testing is confirmed through periodic COT and ACTUATION LOGIC TEST. The CHANNEL CALIBRATION is performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined based on a combination of the channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Thus a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the ΔP error has not changed. An evaluation of extended operation at part power conditions would conclude that it is prudent to administratively adjust the setpoint of the Power Range Neutron Flux (high setpoint) bistables to ≤ 85% RTP when: 1) the power range channel output is adjusted in the decreasing power direction due to a part power calorimetric below 70% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part power conditions would also conclude that the potential need to adjust the indication of the Power Range Neutron Flux in the decreasing power direction is quite small, primarily to address operation in the intermediate range about P-10 (nominally 10% RTP) to allow enabling of the Power Range Neutron Flux (low setpoint) and the Intermediate Range Neutron Flux reactor trips. Before the Power Range Neutron Flux (high setpoint) bistables are reset to ≤ 109% RTP, the power range channel adjustment must be confirmed based on a calorimetric performed at ≥ 70% RTP.

The Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours are allowed for performing the first Surveillance after reaching 15% RTP. A power level of 15% RTP is chosen based on plant stability, i.e. automatic rod control capability and turbine generator synchronized to the grid.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function and Overpower ΔT Function.

A Note clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT. This test shall verify RTB train OPERABILITY by actuation of the two RTBs for each train to their tripped state. Each RTB may be actuated together or individually.

The RTB train test shall include three separate but overlapping tests: (1) The Undervoltage Test for verification RTB operability using only ~~of~~ the undervoltage trip mechanism. (2) The Shunt Trip test for verification of RTB operability using only the~~,~~ shunt trip mechanisms.~~, and~~ (3) ~~t~~The Manual Reactor Trip hardwired switches. The Undervoltage Test shall bypass the shunt trip mechanism, so each RTB actuates using only the undervoltage mechanism. The Shunt Trip Test shall bypass the undervoltage mechanism, so each RTB actuates using only the shunt trip mechanism. The Manual Reactor Trip Test shall actuate the RTB with both mechanisms. Figure 4.4-1 of Topical Report MUAP-07004 (Ref. 6) describes an acceptable overlapping method for conducting these three separate tests that confirms OPERABLE status.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The RTS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 6 and 7. The ACTUATION LOGIC TEST is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function and Overpower ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT.

The RTS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing encompasses all digital Trip Setpoints and trip functions. The self-testing is described in Reference 6 and 7. The COT is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation, including digital Trip Setpoint values or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7.

A COT ensures the entire channel will perform the intended Function. A COT also ensures that the logic processing for interlocks (i.e., P-6 and P-10) is operating correctly. The combination of the COT, CHANNEL CALIBRATION, continuous self-testing and continuous CHANNEL CHECK ensures the comple P-6 and P-10 interlocks are operating correctly.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

The Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTB close for 4 hours this Surveillance must be performed prior to over 4 hours after entry into MODE 3.

SR 3.3.1.8

Performance of the CHANNEL CHECK within 4 hours after reducing power below P-6 and the frequency in accordance with the Surveillance Frequency Control Program ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined based on a combination of the channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 4 hours is based on the need to verify OPERABILITY of the SR instruments within a reasonable time after being re-energized. The Surveillance Frequency thereafter is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

A CHANNEL CHECK may be conducted manually or automatically. For the US-APWR an automated CHANNEL CHECK is normally conducted continuously. Where the CHANNEL CHECK is conducted automatically, an alarm shall be generated when the agreement criteria is not met.

The equipment that performs the automated CHANNEL CHECK, and continuous self-testing described for COT and ACTUATION LOGIC TEST, shall be confirmed OPERABLE including the capability to generate fault alarms.

SR 3.3.1.9

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy as defined by the Allowable Value.

For analog measurements, the CHANNEL CALIBRATION confirms the accuracy of the channel from sensor to ~~analog input module~~digital VDU read out (Ref. 6). The CHANNEL CALIBRATION confirms the analog measurement accuracy conforms to the Allowable Value at multiple points over the entire measurement channel span, encompassing all reactor trip and interlock Trip Setpoint values. Digital reactor trip and interlock Trip Setpoint values are confirmed through a COT.

For binary measurements, the CHANNEL CALIBRATION confirms the accuracy of the channel's state change, as described in Reference 6. The state change must occur within the Allowable Value of the Trip Setpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.13 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety Related Instrumentation."

2. FSAR ~~Chapter 7~~Section 7.2.

3. FSAR Chapter 15.

4. IEEE-603-1991.

5. 10 CFR 50.49.

6. MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), “Safety I&C System Description and Design Process.”

7. MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), “Safety System Digital Platform – MELTAC.”

8. 10 CFR 50.36.

9. FSAR ~~Chapter 6~~Section 6.2.1.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured,

- The RPS provides signal conditioning, analog to digital conversion, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to plant process components, and digital output to control board/main control room/miscellaneous VDUs, and

- The ESFAS and Safety Logic System (SLS) provides Actuation Logic, and Actuation Outputs to initiate the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the partial actuation inputs from the RPS.

The Allowable Value in conjunction with the Trip Setpoint and LCO establishes the threshold for ESFAS action to prevent exceeding acceptable limits such that the consequences of Postulated Accidents (PAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the ~~channel~~measured accuracy is found not to exceed the Allowable Value during the CHANNEL CALIBRATION. The CHANNEL CALIBRATION verifies the instrument at five calibration setpoints corresponding to 0%, 25%, 50%, 75% and 100% of the instrument range. Note that, although a channel is "OPERABLE" under these circumstances, the channel ~~accuracy~~ must be left adjusted to within the established channel calibration tolerance band in accordance with the uncertainty assumptions stated in the referenced setpoint methodology, (as-left criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

BASES

BACKGROUND (continued)

Generally, if a parameter is used for input to the protection circuits and a control function, three channels with a two-out-of-three logic are also sufficient to provide the required reliability and redundancy. The Signal Selection Algorithm (SSA) within the PCMS ensures the control systems can withstand an input failure to the control system without causing erroneous control system operation which would otherwise require the protection function actuation. Since the input failure does not cause an erroneous control system action that challenges the protection function, the input failure is considered a single failure in the ~~RTS~~ESFAS and the ~~RTS~~ESFAS remains capable of providing its protective function with the remaining two operable channels. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-603-1991 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

Allowable Values and ESFAS Setpoints

The Trip Setpoints used in the digital bistables are based on the Analytical Limits defined in the accident analysis and the channel uncertainty. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Allowable Values and Trip Setpoints specified in Table 3.3.2-1 in the accompanying LCO are conservative with respect to the Analytical Limits. The methodology used to calculate the Allowable Values and ESFAS setpoints incorporates all of the known uncertainties applicable to each channel (Ref. 7). The magnitudes of these uncertainties are factored into the determination of each ESFAS Trip Setpoint and corresponding Allowable Value. The ESFAS Trip Setpoint entered into the digital bistable is more conservative than that specified by the Analytical Limit to account for measurement errors detectable by the CHANNEL CALIBRATION. The Allowable Value serves as the Technical Specification OPERABILITY limit for the purpose of the CHANNEL CALIBRATION. One example of such a change in measurement error is drift during the surveillance interval. If the measured accuracy does not exceed the Allowable Value, the channel is considered OPERABLE.

BASES

BACKGROUND (continued)

The digital bistable outputs from all trains of the RPS are sensed by each ESFAS train and combined into logic that represent combinations indicative of various transients. If a required logic combination is completed, the ESFAS train will send actuation signals via the Safety Bus to its respective SLS train. The SLS actuates those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

The ESFAS and SLS are continuously automatically sel-tested while the unit is at power. When any one train is taken out of service for manual testing, the remaining trains are capable of providing unit monitoring and protection until the testing has been completed.

The actuation of ESF components is accomplished through solid state Actuation Outputs. The SLS energizes the Actuation Outputs appropriate for the condition of the unit. Each Actuation Output energizes one plant component. Actuation Outputs are tested in conjunction with their respective plant components. This test overlaps with the continuous automatic self-testing.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Low Pressurizer Pressure is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3~~ and 10~~).

The LCO requires all instrumentation performing an ESFAS Function, listed in Table 3.3.2-1 in the accompanying LCO, to be OPERABLE. A channel is OPERABLE provided the "as-found" accuracy value does not exceed its associated Allowable Value. A trip setpoint may be set more conservative than the Trip Setpoint as necessary in response to plant conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

These other functions ensure:

- Isolation of nonessential systems through containment penetrations,

- Trip of the reactor to limit power generation,

- Isolation of main feedwater (MFW) to limit secondary side mass losses,

- Start of EFW to ensure secondary side cooling capability,

- Isolation of the main control room to ensure habitability, and

- Trip of the Reactor Coolant Pump to prevent the unexpected Reactor Coolant Pump Trip after a small break LOCA.

a. ECCS Actuation - Manual Initiation

The LCO requires three trains to be OPERABLE. The operator can initiate ECCS at any time by using any two out of four ECCS - Manual Initiation switches in the main control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each train consists of one push button and the interconnecting wiring to the actuation logic cabinet. Each push button actuates its own train directly. A signal from each pushbutton is also interfaced to all other trains via internal PSMS communication links. In addition to direct actuation by its own train pushbutton, each train is also actuated by two out of three Manual Initiation signals received from the other trains.

b. ECCS Actuation - Actuation Logic and Actuation Outputs

This LCO requires three trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the actuation output devices responsible for actuating the ESF equipment.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Low Main Steam Line Pressure must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic ECCS actuation via High Containment Pressure, and outside containment SLB will be terminated by the High Main Steam Line Pressure Negative Rate signal for main steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

2. Containment Spray

Containment Spray provides ~~three~~two primary functions:

1. Lowers containment pressure and temperature after an HELB in containment, and

2. Reduces the amount of radioactive iodine in the containment atmosphere.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure,
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure, and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Containment spray is actuated manually or by High 3 Containment Pressure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

For any single containment penetration, isolation can be accomplished by either of two redundant trains. However, all containment isolation functions are distributed among all four ESFAS trains.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW), at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by ECCS Actuation, or manually via the Actuation Logic. All process lines penetrating containment, with the exception of CCW, are isolated.

CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by ~~four~~two switches in the main control room. Each push button actuates its own train directly. A signal from each pushbutton is also interfaced to all other trains via internal PSMS communication links. In addition to direct actuation by its own train pushbutton, each train is also actuated by two out of three Manual Initiation signals received from the other trains.

Note that manual actuation of Phase A Containment Isolation also actuates Containment Purge Isolation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. Main Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the main steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the main steam lines depressurize. ~~For units that do not have main steam line check valves,~~ Main Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven EFW pump during a feed line break.

Main Steam Line Isolation components are distributed to Trains A and D.

a. Main Steam Line Isolation - Manual Initiation

Manual initiation of Main Steam Line Isolation can be accomplished from the main control room. There are two switches in the main control room, one for each train. Either switch can initiate action to immediately close all MSIVs. The LCO requires two trains to be OPERABLE.

b. Main Steam Line Isolation - Actuation Logic and Actuation Outputs

Actuation Logic and actuation outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Main Steam Line Isolation valves are distributed to Trains A and D. Both trains must be OPERABLE.

Manual and automatic initiation of Main Steam Line Isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Main Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed ~~and de-activated~~. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Main Steam Line Isolation - High-High Containment Pressure

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. High-High Containment Pressure provides no input to any control functions. There are four High-High Containment Pressure channels in a two-out-of-four logic configuration. Three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties.

High-High Containment Pressure must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Main Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the High-High Containment Pressure setpoint.

d. Main Steam Line Isolation - Main Steam Line Pressure

(1) Low Main Steam Line Pressure

Low Main Steam Line Pressure provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven EFW pump. Low Main Steam Line Pressure was discussed previously under ECCS Function 1.e~~.1~~.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Low Main Steam Line Pressure Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via High-High Containment Pressure. Stuck valve transients and outside containment SLBs will be terminated by the High Main Steam Line Pressure Negative Rate signal for Steam Line Isolation below P-11 when ECCS has been manually blocked. The Main Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed ~~and de-activated~~. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) High Main Steam Line Pressure Negative Rate

High Main Steam Line Pressure Negative Rate provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Low Main Steam Line Pressure main steam isolation signal when less than the P-11 setpoint, the High Main Steam Line Pressure Negative Rate signal is automatically enabled. High Main Steam Line Pressure Negative Rate provides no input to any control functions. There are four High Main Steam Line Pressure Negative Rate signals in a two-out-of-four logic configuration. Three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

High Main Steam Line Pressure Negative Rate must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the main steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Low Main Steam Line Pressure signal is automatically enabled. The Main Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

5. Main Feedwater Isolation

5A. Main Feedwater ~~Control~~Regulation Valve Closure

The primary function of the Main Feedwater ~~Control~~Regulation Valve Closure is to stop the excessive flow of feedwater into the SGs. This Function is necessary to mitigate the effects of a high water level in the SGs, which could result in excessive cooldown of the primary system.

This Function is actuated when T_{avg} is less than the low setpoint coincident with reactor trip, and closes all the main Feedwater ~~control~~Regulation valves.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Main Feedwater Isolation - Low T_{avg}

~~Low~~ There are four Low T_{avg} channels per loop in a two out of four configuration. Three channels of T_{avg} per loop are required to be OPERABLE. The T_{avg} channels are combined in a logic such that two out of three channels tripped cause a trip for the parameter. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels in a two-out-of-four configuration ensures no single random failure disables the Low T_{avg} Function. The T_{avg} channels provide control inputs interfaced from the PSMS to the PCMS through an SSA. But the control function cannot initiate events that the Function acts to mitigate. Therefore, the SSA is not required to address control protection interaction issues.

With the T_{avg} resistance temperature detectors (RTDs) located inside the containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrumental uncertainties.

The Main Feedwater Isolation - Low T_{avg} signal is enabled by the Main Feedwater Isolation - Reactor Trip, P-4 interlock, described below.

Coincident with Reactor Trip, P-4

The Main Feedwater Isolation - Low T_{avg} signal is enabled when the reactor is tripped as indicated by the P-4 interlock. Therefore, the requirements for the P-4 interlock are not repeated in Table 3.3.2-1. Instead, Function 11, Reactor Trip P-4, is referenced for the initiating Function and requirements. Note that all four Turbine Trip actuation trains are actuated when any two out of four RTB trains are actuated.

5B. Main Feedwater Isolation

The primary function of the Main Feedwater Isolation is to stop the excessive flow of feedwater into the SGs. This Function is necessary to mitigate the effects of a high water level in the SGs, which could result in excessive cooldown of the primary system. The High SG Water Level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high-high setpoint, and performs the following functions:

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Trips the MFW pumps,
- Initiates feedwater isolation, and
- Shuts the MFW regulating valves, the MFW bypass feedwater regulating valves and the SG water filling control valves.

This Function is actuated by High-High SG Water Level or by an ECCS Actuation signal or manually. The ECCS Actuation signal was discussed previously.

Main Feedwater Isolation components are distributed to Trains A and D.

a. Main Feedwater Isolation - Manual Initiation

Manual initiation of Main Feedwater Isolation can be accomplished from the main control room. There are two switches in the main control room, one for each train. Either switch can initiate action to immediately close all feedwater isolation valves. The LCO requires two trains to be OPERABLE.

b. Main Feedwater Isolation - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Main Feedwater isolation valves are distributed to Trains A and D. Both trains must be OPERABLE.

c. Main Feedwater Isolation - High High Steam Generator Water Level ~~(P-14)~~

This signal provides protection against excessive feedwater flow. There are four High High Steam Generator Water Level channels in a two-out-of-four logic configuration for each Steam Generator. The ESFAS SG water level instruments provide input to the SG Water Level Control System. The interface from the safety channels in the PSMS to the PCMS is through the Signal Selector Algorithm (SSA). The SSA ensures an input failure to the control system does not result in erroneous control system action that would require the protection function actuation. Therefore, the protection function requires only two additional channels to provide the protection function actuation. Three channels total must be OPERABLE.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

d. Main Feedwater Isolation - ECCS Actuation

Main Feedwater Isolation is also initiated by all Functions that initiate ECCS Actuation. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their ECCS Actuation function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, ECCS Actuation, is referenced for all initiating functions and requirements. Note that both Main Feedwater Isolation trains are actuated when any two out of four ECCS Actuation - Automatic or Manual Initiation signals are actuated.

All Main Feedwater Isolation Functions except for High-High Steam Generator Water Level must be OPERABLE in MODES 1 and 2 and 3 except when all MFIVs, MFRVs, MFBRVs and SGWFCVs ~~and associated bypass valves~~ are closed when the MFW System is in operation. In MODES 4, 5, and 6, the MFW System is not in service and this Function is not required to be OPERABLE.

High-High Steam Generator Water Level must be OPERABLE in MODES 1 and 2 and 3 (above P-11) except when all MFIVs, MFRVs, MFBRVs and SGWFCVs are closed when the MFW System is in operation. This signal may be manually blocked by the operator below the P-11 setpoint. In MODES 4, 5, and 6, the MFW System is not in service and this Function is not required to be OPERABLE.

6. Emergency Feedwater Actuation

The EFW Actuation System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has four trains, with two motor driven pumps and two turbine driven pumps, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe

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function actuation. Three channels total must be OPERABLE.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

d. Emergency Feedwater Actuation - ECCS Actuation

An ECCS Actuation signals all four EFW trains. The EFW initiation functions are the same as the requirements for their ECCS Actuation function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, ECCS Actuation, is referenced for all initiating functions and requirements.

e. Emergency Feedwater Actuation - Loss of Offsite Power

A loss of offsite power will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each Class 1E bus (4 trains). The voltage drop is detected by three undervoltage devices on each bus, in a two out of three configuration. Loss of power to a Class 1E bus will actuate its respective EFW train (with either its motor or turbine driven pump). ~~to~~This ensures that, for a sitewide loss of offsite power, at least two SGs contain~~s~~ enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

The LCO requires ~~two~~three OPERABLE undervoltage devices on each Class 1E bus corresponding to each OPERABLE EFW train.

Functions 6.a through 6.~~d~~e must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. Low SG Water Level in any operating SG will cause the EFW trains to actuate. The system is aligned so that upon a start of the EFW pump, water immediately begins to flow to the SGs. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

generated in the reactor to require the SGs as a heat sink. In MODE 4, EFW actuation does not need to be OPERABLE because either EFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

f. Emergency Feedwater Actuation - Trip of All Main Feedwater Pumps

A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. Each motor driven MFW pump is equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Emergency Feedwater Actuation on trip of all main feedwater pumps is an anticipatory function that is not credited in the safety analysis. Therefore, the LCO requires one OPERABLE channel per pump. A trip of all MFW pumps actuates all EFW trains to ensure that at least two SGs are available with water to act as the heat sink for the reactor.

This function must be OPERABLE in MODES 1 and 2. This ensures that at least two SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the MFW pumps may be normally shut down, and thus neither pump trip is indicative of a condition requiring automatic EFW initiation.

7. Emergency Feedwater Isolation

One of the objectives of EFW Isolation is to prevent SG overfill in the event of SGTR. The Other objective of EFW Isolation is to stop the flow of EFW into the affected SG in the event of MSLB. For both objectives, the EFW isolation Functions are automatically actuated by High SG Water Level signal, or by Low Main Steam Line Pressure signal. The Function may also be actuated manually. EFW Isolation is distributed to Trains A and D.

a. Emergency Feedwater Isolation - Manual Initiation

This LCO requires 2 Manual EFW Isolation Actuation trains for each SG. This Function closes the EFW Isolation Valve for the SG associated with the switches.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Emergency Feedwater Isolation - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Emergency Feedwater isolation valves are distributed to Trains A and D. Trains A and D must be OPERABLE.

Manual and automatic initiation of EFW Isolation Functions must be OPERABLE in MODES 1, 2, and 3 when the SGs are in operation. In MODES 4, 5, and 6, SGs are not in service and this Function is not require to be OPERABLE.

c. Emergency Feedwater Isolation - High Steam Generator Water Level coincident with P-4 signal and no Low Main Steam Line Pressure

This signal provides protection against damaged SG overfill. There are four High Steam Generator Water Level channels in a two-out-of-four logic configuration for each Steam Generator. The ESFAS SG water level instruments provide input to the SG Water Level Control System. The interface from the safety channels in the PSMS to the PCMS is through the Signal Selector Algorithm (SSA). The SSA ensures an input failure to the control system does not result in erroneous control system action that would require the protection function actuation. Therefore, the protection function requires only two additional channels to provide the protection function actuation. Three channels total must be OPERABLE.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

High Steam Generator Water Level must be OPERABLE in MODES 1, 2 and 3 (above P-11) when the SGs are in operation. This signal may be manually blocked by the operator below the P-11 setpoint. This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. In MODES 4, 5, and 6, SGs

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

are not in service and this Function is not required to be OPERABLE.

d. Emergency Feedwater Isolation - Low Main Steam Line Pressure

This signal provides protection against excessive cooling from damaged SG. A steam line break or a feed line brake inside of containment, would result in a low steam line pressure.

Low Steam Line Pressure provides no input to any control functions. There are four Low Main Steam Line Pressure channels on each steam line in a two-out-of-four logic configuration. Three OPERABLE channels on each main steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

~~EFW Isolation Functions must be OPERABLE in MODES 1, 2 and 3 when the SG is in operation. In MODES 4, 5, and 6, SGs are not in service and this Function is not required to be OPERABLE.~~Low Main Steam Line Pressure must be OPERABLE in MODES 1, 2 and 3 (above P-11) when the SGs are in operation. This signal may be manually blocked by the operator below the P-11 setpoint. This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. In MODES 4, 5, and 6, SGs are not in service and this Function is not required to be OPERABLE.

8. CVCS Isolation

The objective of CVCS Isolation is to prevent Pressurizer overfill in the event of a CVCS malfunction. For this objective, the CVCS Isolation is automatically actuated by High Pressurizer Water Level signal. The Function may also be actuated manually.

CVCS Isolation valves are distributed to Trains A and D. Both trains must be OPERABLE.

a. CVCS Isolation – Manual Initiation

Manual initiation of CVCS Isolation can be accomplished from the main control room. There are two switches in the main control room, one for each train. This LCO requires 2 Manual CVCS Isolation Actuation

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. CVCS Isolation – Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. CVCS Isolation valves are distributed to Trains A and D. Both trains must be OPERABLE.

Manual and automatic initiation of CVCS Isolation Functions must be OPERABLE in MODES 1, 2 and 3. In MODES 4, 5, and 6, the Pressurizer may be filled with water and this Function is not required to be OPERABLE.

c. CVCS Isolation - High Pressurizer Water Level

This signal provides protection against that the Pressurizer overfill in the event of CVCS malfunction.

There are four High Pressurizer Water Level channels in a two-out-of-four logic configuration. Pressurizer Water Level provides input to the Pressurizer Level Control System. The interface from the safety channels in the PSMS to the PCMS is through the Signal Selector Algorithm (SSA). The SSA ensures an input failure to the control system does not result in erroneous control system action that would require the protection function actuation. Therefore, the protection function requires only two additional channels to provide the protection function actuation. Three channels total must be OPERABLE.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

High Pressurizer Water Level must be OPERABLE in MODES 1, 2 and 3 (above P-11). This signal may be manually blocked by the operator below the P-11 setpoint. This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. In MODES 4, 5, and 6, the Pressurizer may be filled with water and this Function is not required to be OPERABLE.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~CVCS Isolation Functions must be OPERABLE in MODES 1, 2 and 3. In MODES 4, 5, and 6, the Pressurizer may be filled with water and this Function is not required to be OPERABLE.~~

9. Turbine Trip

The primary functions of the Turbine Trip are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive cooldown of the primary system.

The Turbine Trip Function is actuated by High-High Steam Generator Water Level ~~(P-14)~~ or on Reactor Trip (P-4).

a. Turbine Trip - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Turbine trip solenoid valves are arranged in a selective one out of two twice configuration. A turbine trip will occur from Turbine Trip actuation from Trains A or C, and Trains B or D. The LCO requires all four trains to be OPERABLE.

b. Turbine Trip - Reactor Trip, P-4

The turbine is tripped on a reactor trip. Turbine trip on reactor trip is an un-credited non-safety function in the safety analysis. However, turbine trip on reactor trip is assumed in the safety analysis in order to prevent unnecessary ECCS Actuation and to shift to the safe shutdown state by appropriate actions after AOO and PA conditions.

Turbine Trip is initiated when the reactor trips as indicated by the P-4 interlock. Therefore, the requirements for the P-4 interlock are not repeated in Table 3.3.2-1. Instead, Function 11, Reactor Trip P-4, is referenced for the initiating Function and requirements. Note that all four Turbine Trip actuation trains are actuated when any two out of four RTB trains are actuated.

c. High-High Steam Generator Water Level

The High-High Steam Generator Water Level signal prevents water in the steam lines that could lead to turbine generator damage. Turbine trip on High-High Steam Generator Water Level is an un-credited non-safety function in the safety analysis.

There are four High-High Steam Generator Water Level

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

channels in a two-out-of-four logic configuration for each Steam Generator. The PSMS SG water level instruments provide input to the SG Water Level Control System. The interface from the safety channels in the PSMS to the PCMS is through the Signal Selector Algorithm (SSA). The SSA ensures an input failure to the control system does not result in erroneous control system action that would require the protection function actuation. Therefore, the protection function requires only two additional channels to provide the protection function actuation. Three channels total must be OPERABLE.

10. Reactor Coolant Pump Trip

TMI Action Plan Item II.K.3.5 (Ref. 1) requires automatic trip of reactor coolant pumps (~~PCPs~~RCPs) following a loss-of-coolant accident (LOCA). The requirement is based on the consideration that a delayed-trip or continuous operation of the RCPs during a small break LOCA would lead to more severe consequences than if the RCPs are tripped early following a postulated break. Tripping all the RCPs early during a small break LOCA precludes the occurrence of excessive fuel cladding temperature.

a. Reactor Coolant Pump Trip – ECCS Actuation coincident with P-4 signal

The consequence of continuous RCP operation is the extensive liquid discharge from the break beyond the time that the system would drained down to allow steam discharge from the break had the pumps been immediately tripped. Therefore pump trip following a reactor trip and indication of ECCS actuation would be effective.

For the small break LOCA analysis, the loss of offsite power triggered by reactor trip signal is conservatively assumed, which would cause the earliest RCP trip. In case that the automatic RCP trip is enabled, an earlier RCP trip results in earlier flow coastdown leading to more severe consequences.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocksare included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when RTBs have opened in two out of four RTB trains. RTB position signals from each RTB are interfaced to all PSMS trains via internal PSMS data links so that the P-4 interlock is generated independently within each train. Therefore this LCO requires three trains to be OPERABLE.

This Function allows operators to take manual control of ECCS systems after the initial phase of ECCS Actuation is complete. Once ECCS is overridden, automatic actuation of ECCS cannot occur again until the RTBs have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate MFW with coincident low T_{avg},
- Enable a manual override of ECCS Actuation and prevent ECCS reactuation,
- EFW Isolation with coincident High SG Water level and no Low Main Steam Line Pressure, and
- Trip the Reactor Coolant Pump with coincident ECCS Actuation.

Each of the above Functions except Reactor Coolant Pump Trip is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Pressure ECCS Actuation signals, the Low Main Steam Line Pressure main steam line isolation signal, the CVCS Isolation signal, the EFW Isolation signals, and the High-High SG Water Level Main Feedwater Isolation signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons. When the Low Main Steam Line Pressure main steam line isolation signal is enabled, the main steam isolation on High Main Steam Line Pressure Negative Rate is disabled. The Trip Setpoint reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of ECCS, main steam line isolation, CVCS Isolation, EFW Isolation or Main Feedwater Isolation on High-High SG Water Level. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

12. Containment Purge Isolation

Containment Purge Isolation initiates on ~~Containment Radiation~~Containment High Range Area Radiation, an automatic ECCS Actuation signal, by manual initiation of Containment Isolation Phase A, or by manual initiation of Containment Spray.Containment Purge Isolation components are distributed to PSMS Trains A and D. Two trains are sufficient to provide the safety function. Both are required to be OPERABLE to provide the safety function with a concurrent single failure.

a. Containment Isolation Phase A - Manual Initiation

Containment Purge Isolation is manually initiated by Containment Isolation Phase A - Manual Initiation. The Containment Purge Isolation requirements for this Function are the same as the requirements for the Containment Isolation Phase A Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 3.a, Containment Isolation Phase A-Manual Initiation, is referenced for all initiating Functions and requirements.

b. Containment Spray - Manual Initiation

Containment Purge Isolation is manually initiated by

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Containment Spray - Manual Initiation. The Containment Purge Isolation requirements for this Function are the same as the requirements for the Containment Spray Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 2.a, Containment Spray – Manual Initiation, is referenced for all initiating Functions and requirements.

Note that all ~~four~~two Containment Purge Isolation trains are actuated when any two-out-of-four Containment Spray - Manual Initiation signals are actuated.

c. Containment Purge Isolation - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Containment Purge Isolation valves are distributed to Trains A and ~~B~~D. Both trains must be OPERABLE.

d. Containment Purge Isolation - ECCS Actuation

Containment Purge Isolation is also initiated by all Functions that initiate ECCS. The Containment Purge Isolation requirements for these Functions are the same as the requirements for the ECCS function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, ECCS, is referenced for all initiating Functions and requirements.

Note that all ~~four~~two Containment Purge Isolation trains are actuated when any two out of four ECCS - Automatic or Manual Initiation signals are actuated.

e. Containment Purge Isolation - ~~Containment Radiation~~Containment High Range Area Radiation

~~Containment Radiation~~Containment High Range Area Radiation has four channels in a two-out-of-four logic configuration. Three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic.

The Containment Purge Isolation Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Purge Isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the Containment ~~Ventilation~~Purge Isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within acceptable limits.

13. Main Control Room Isolation

The Main Control Room Isolation function provides an enclosed main control room environment from which the unit can be operated following an uncontrolled release of radioactivity. MCR Isolation controls the main control room HVAC System (MCRVS) which includes two subsytems: ~~Main Control Room Intake Duct Isolation~~Main Control Room Emergency Filtration System (MCREFS) and ~~Main Control Room Emergency Recirculation Fan.~~Main Control Room Air Temperature Control System (MCRATCS), described in the DCD Chapter 16 Section 3.7.10

There are four Main Control Room Isolation trains. Train A and D of MCR Isolation control two 100% capacity trains of subsystem MCREFS, and all four trains of MCR Isolation control four 50% capacity trains of subsystem MCRATCS. ~~Three~~Two trains of MCR Isolation, including A or D, must actuate to properly provide the safety function (i.e., isolate and supply filtered air to the main control room) and~~. Four~~three trains, including A and D, must be OPERABLE to provide the safety function with a concurrent single failure.

The MCR Isolation actuation instrumentation consists of redundant radiation monitors. A high radiation signal will initiate all four MCR Isolation trains. The main control room operator can also initiate MCR Isolation trains by manual switches in the main control room. MCR Isolation is also actuated by an ECCS Actuation signal.

The main control room must be kept habitable for the operators stationed there during accident recovery and post accident operations. The MCR Isolation function acts to terminate the supply of unfiltered outside air to the main control room, initiate filtration, and allows pressurization of the main control room. These actions are necessary to ensure the main control room is kept habitable for the operators stationed there during

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

accident recovery and post accident operations by minimizing the radiation exposure of the main control room personnel. In MODES 1, 2, 3, and 4, the radiation monitor actuation of MCR Isolation is a backup for the ECCS Actuation. This ensures initiation of the MCR Isolation during a loss of coolant accident or steam generator tube rupture.

The radiation monitor actuation of MCR Isolation during movement of irradiated fuel assemblies are the primary means to ensure main control room habitability in the event of a fuel handling accident.

a. Manual Initiation

The LCO requires four trains OPERABLE. The operator can initiate MCR Isolation for all four MCR Isolation trains at any time by using four Manual Initiation switches in the main control room. Each push button actuates its own train directly. A signal from each pushbutton is also interfaced to all other trains via internal PSMS communication links. In addition to direct actuation by its own train pushbutton, each train is also actuated by two-out-of-three Manual Initiation signals received from the other trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one push button and the interconnecting wiring to the ESFAS cabinet.

b. Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b., ECCS Actuation. However, for MCR Isolation 4 trains must be operable due to the distribution of main control room isolation dampers to all four trains.

c. Main Control Room Radiation

There are three kinds of Main Control Room Radiation monitor functions (gas monitor, iodine monitor, and particulate monitor). One monitor includes two detectors of train A and D. RPS trains A and D provide separate bistable setpoint comparison functions for each

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

monitor. These bistable output signals are distributed from RPS trains A and D to each of the four ESFAS trains. Within each of the four ESFAS trains the MCR Isolation signal is actuated on a signal from either the A or D train detectors using 1-out-of-2 logic for each type of monitor. ~~A signal from either monitor for each function will actuate all four MCR Isolation trains.~~

The LCO specifies two required Main Control Room Radiation monitors for each function to ensure that the radiation monitoring instrumentation necessary to initiate the MCR Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

d. ECCS ACTUATION

Main Control Room Isolation is also initiated by all Functions that initiate ECCS Actuation. The MCR Isolation requirements for these Functions are the same as the requirements for their ECCS Actuation function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, ECCS Actuation, is referenced for all initiating Functions and requirements. Note that all four MCR Isolation trains are actuated when any two out of four ECCS Actuation - Automatic or Manual Initiation signals are actuated.

The MCR Isolation Functions must be OPERABLE in MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies.

14. Block Turbine Bypass and Cooldown Valves

The Block Turbine Bypass and Cooldown Valves function prevents the overcooldown of the reactor coolant system when Tavg is decreased abnormally.

Block turbine bypass and cooldown valves are distributed to Trains A and D. Both trains must be OPERABLE.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Block Turbine Bypass and Cooldown Valves - Manual Initiation

Manual initiation of Block Turbine Bypass and Cooldown Valves can be accomplished from the main control room. There are two switches in the main control room, one for each train. This LCO requires 2 Manual Block Turbine Bypass and Cooldown Valves Actuation switches. Operation of either switch will actuate this Function.

b. Block Turbine Bypass and Cooldown Valves - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Block turbine bypass and cooldown valves are distributed to Trains A and D. Both trains must be OPERABLE.

c. Block Turbine Bypass and Cooldown Valves - Low-low T_{avg} Signal

This function must be OPERABLE in MODES 1, 2 and 3. In MODES 4, 5, and 6, the average coolant temperature is below the low-low Tavg signal setpoint and this function is not required to be OPERABLE.

The ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 9).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's accuracy is found non-conservative with respect to the Allowable Value, or the transmitter, instrument Loop, signal processing electronics, or digital bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

ACTIONS (continued)

The initial completion time of 72 hours is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19. The PSMS reliability analysis credits the continued compliance to the single failure criteria, since the ESFAS manual initiation function remains fully operable from the Safety VDUs, even when one ESFAS manual initiation function is inoperable.

C.1, C.2.1, and C.2.2

Condition C applies to the Actuation Logic and Actuation Outputs for the following functions:

- Containment Phase A Isolation, and
- Containment Phase B Isolation.

This action addresses the train orientation of the PSMS. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is reasonable considering that there are sufficient trains OPERABLE to ensure the capability of the required Function, and the low probability of an event occurring during this interval. The Completion Time also considers that the remaining OPERABLE trains each have automatic self-testing.

If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (30 hours total time) and in MODE 5 within an additional 30 hours (60 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train(s) is OPERABLE. This allowance is based on the reliability analysis assumption that 4 hours is the average time required to perform train surveillance.

The bypassed condition for up to 4 hours and the initial completion time of 24 hours are justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

ACTIONS (continued)

D.1, D.2.1, and D.2.2

Condition D applies to:

- High Containment Pressure,
- Low Pressurizer Pressure,
- Low Main Steam Line Pressure,
- Low T_{avg},
- High Pressurizer Water Level,
- High-High Containment Pressure,
- High Main Steam Line Pressure Negative Rate,
- High SG Water Level,
- Low SG Water Level, ~~and~~
- High-High SG Water Level~~.~~, and
- Low-low T_{avg}

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-three configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified because the remaining two OPERABLE channels have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

BASES

ACTIONS (continued)

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

~~The Required Actions are modified by a Note that allows an inoperable Containment Pressure or Main Steam Line Pressure channel to be bypassed for up to 12 hours for surveillance testing of other channels.~~One channel may be bypassed for up to 12 hours for surveillance testing.

One channel may be bypassed for up to 12 hours for surveillance testing. The 12 hours bypass limit is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19. This bypass is not allowed for the other functions because these channels are also used for control. If a failure were to occur in one of the two remaining control channels, a plant transient could occur that would require a plant trip, but a plant trip would not occur with only one remaining operable channel.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray - High-3 Containment Pressure, and
- Containment Phase B Isolation - High-3 Containment Pressure.

This LCO requires three operable channels. None of these signals has input to a control function. Two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented.

Restoring the channel to OPERABLE status within 72 hours is sufficient because of the low probability of an event occurring during this interval because the remaining two OPERABLE channels have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS.

ACTIONS (continued)

Failure to restore the required number of channels to OPERABLE status within 72 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

F.1, F.2.1, and F.2.2

Condition F applies to:

- Loss of Offsite Power, and
- P-4 Interlock~~,~~.

Condition F also applies to the manual initiation for:

- Main Steam Line Isolation,
- Main Feedwater Isolation,
- Emergency Feedwater Actuation,
- Emergency Feedwater Isolation, ~~and~~
- CVCS Isolation~~.~~, and
- Block Turbine Bypass and Cooldown Valves.

For all Functions, this action addresses the train orientation of the PSMS. For the Loss of Offsite Power Function, this action also recognizes the lack of manual trip provision for a failed channel. If a train or channel is inoperable, 72 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval.

BASES

ACTIONS (continued)

If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

The initial completion of 72 hours is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19. For the manual initiation functions, the PSMS reliability analysis credits the continued compliance to the single failure criteria, since the ESFAS manual initiation function remains fully operable from the Safety VDUs, even when one ESFAS manual initiation function is inoperable.

G.1, G.2.1, and G.2.2

Condition G applies to the Actuation Logic and Actuation Outputs for the;

- Emergency Feedwater Isolation,
- CVCS Isolation, and
- Turbine Trip Functions.

The action addresses the train orientation of the PSMS for these functions. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is reasonable considering that the safety function can be performed by the remaining OPERABLE trains, and the low probability of an event occurring during this interval. The Completion Time also considers that the remaining OPERABLE trains each have continuous self-testing.

If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

BASES

ACTIONS (continued)

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other trains are OPERABLE. This allowance is based on the assumption that 4 hours is the average time required to perform channel surveillance.

The bypassed condition for up to 4 hours and the initial completion time of 24 hours are justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

H.1 and H.2

Condition H applies to the EFW pump start on trip of all MFW pumps.

This action addresses the train orientation of the PSMS for the auto start function of the EFW System on loss of all MFW pumps. The OPERABILITY of the EFW System must be assured by allowing automatic start of the EFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified because trip of all main feedwater pumps is an anticipatory function that is not credited in the safety analysis.

I.1, I.2.1, and I.2.2

Condition I applies to the P-11 interlock.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of this interlocks.

BASES

ACTIONS (continued)

J.1 and J.2

Condition J applies to the Actuation Logic and Actuation Outputs for the Emergency Feedwater Actuation.

The action addresses the train orientation of the PSMS for these functions. If one train is inoperable, 72 hours are allowed to restore the train to OPERABLE status. The 72 hours allowed for restoring the inoperable train to OPERABLE status is reasonable considering that the safety function can be performed by the remaining OPERABLE trains, and the low probability of an event occurring during this interval. The Completion Time also considers that the remaining OPERABLE trains each have continuous self-testing. Required Action J.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other trains are OPERABLE. This allowance is based on the assumption that 4 hours is the average time required to perform channel surveillance.

The bypassed condition for up to 4 hours and the initial completion time of 72 hours are justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

K.1

Condition K applies to the failure of one ~~Containment Radiation~~Containment High Range Area Radiation monitor channel. Since the three Containment Radiation monitors measure the same parameters, failure of a single channel does not result in loss of the radiation monitoring Function for any events.

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-three configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified because the remaining two OPERABLE channels have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS.

BASES

ACTIONS (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 5 within the next 30 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, these Functions are no longer required OPERABLE.

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

L.1

Condition L applies to the Containment Purge Isolation - Actuation Logic and Actuation Output Function and addresses the train orientation of the Engineered Safety Features Actuation System (ESFAS). It also addresses the failure of multiple Containment Purge Isolation - Containment Radiation Monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action K.1.

If an Actuation Logic and Actuation Output train is inoperable, multiple Containment Radiation Monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition K are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

M.1

Condition M applies to the Actuation Logic and Actuation Outputs Function of the MCR Isolation, the Main Control Air monitor Functions, and the Manual Initiation Functions.

If one Actuation Logic and Actuation Outputs train is inoperable, ~~or~~ one Main Control Room Radiation channel is inoperable in one or more Functions, or one Manual Initiation train is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one train of the affected subsystem(s)~~MCR Isolation train~~ must be placed in the emergency

ACTIONS (continued)

~~radiation protection~~ mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

Affected subsystems depend on inoperable train, as follows.

- If train A or D is inoperable, MCREFS doesn't satisfy the single failure criterion. Therefore, one train MCREFS is placed on emergency mode. MCRATCS is unaffected, since three required trains remain operable.
- If train B or C is inoperable, MCREFS is unaffected and three required trains of MCRATCS remain operable. Therefore, no action is required.

N.1.1, N.1.2, and N.2

Condition N applies to the failure of two MCR Isolation Actuation Logic and Actuation Outputs trains, two Main Control Room Radiation channels, or two Manual Initiation trains for one or more Functions. The first Required Action is to place the affected subsystem(s)~~two OPERABLE MCR Isolation trains~~ in the emergency mode of operation immediately. For MCREFS this requires one train, since each is 100% capacity. Two trains of MCRATCS are required since each is 50% capacity. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the ~~MCR Isolation~~MCRVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, all trains of the affected subsystem(s)may be placed in the emergency mode. This ensures the MCR Isolation function is performed even in the presence of a single failure. ~~The Required Actions are modified by a Note that excludes this alternative for failure of the Actuation Logic and Actuation Outputs, since a failure of this Function affects normal and emergency modes. This alternative is applicable for failure of the Main Control Room Radiation monitor functions and failure of the Manual Initiation function.~~

Affected subsystems depend on inoperable train, as follows.

- If trains A and D are inoperable, MCREFS is completely inoperable. Therefore, one train of MCREFS is placed on emergency mode and the required action of MCRVS is appled (to restore in 7 days). Or two trains of MCREFS are placed on emergency mode. And one train of MCRATCS is placed on emergency mode, since MCRATCS does not satisfy the single failure criterion.

ACTIONS (continued)

- If trains A and B, or A and C, or B and D, or C and D are inoperable, one train of MCREFS and one train of MCRATCS is placed on emergency mode since both subsystems don't satisfy the single failure criterion.

- If trains B and C are inoperable, MCREFS is unaffected. One train of MCRATCS is placed on emergency mode since MCRATCS does not satisfy the single failure criterion.

O.1 and O.2

Condition O applies when the Required Action and associated Completion Time for Condition M or N have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

P.1

Condition P applies when the Required Action and associated Completion Time for Condition M or N have not been met when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require MCR Isolation actuation.

Q.1 and Q.2

Condition Q applies to the Actuation Logic and Actuation Outputs for the following functions:

- ECCS Actuation, and

- Containment Spray.

This action addresses the train orientation of the PSMS. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is reasonable considering that there are sufficient trains OPERABLE to ensure the capability of the required Function, and the low probability of an event occurring during this interval. The Completion Time also considers that the remaining OPERABLE trains each have automatic self-testing. Required Action Q.2 allows the option to apply the requirements of Specification 5.5.18

ACTIONS (continued)

to determine a Risk Informed Completion Time. This Required Action is not applicable in MODE 4.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train(s) is OPERABLE. This allowance is based on the reliability analysis assumption that 4 hours is the average time required to perform train surveillance.

The bypassed condition for up to 4 hours and the initial completion time of 24 hours are justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

R.1 and R.2

Condition R applies to the Actuation Logic and Actuation Outputs for the following functions:

- ECCS Actuation, and
- Containment Spray,

If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 6 hours and in MODE 5 within an additional 30 hours (36 hours total time). The 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.

S.1 and S.2

Condition S applies to the Actuation Logic and Actuation Outputs for the;

- Main Steam Line Isolation, ~~and~~
- Main Feedwater Isolation~~.~~, and
- Block Turbine Bypass and Cooldown Valves.

The action addresses the train orientation of the PSMS for these functions. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is reasonable considering that the safety function can be performed by the remaining OPERABLE trains, and the low probability of an event occurring during this interval. The Completion Time also consider

BASES

ACTIONS (continued)

that the remaining OPERABLE trains each have continuous self-testing. Required Action S.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other trains are OPERABLE. This allowance is based on the assumption that 4 hours is the average time required to perform channel surveillance.

The bypassed condition for up to 4 hours and the initial completion time of 24 hours are justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

T.1 and T.2

Condition T applies to the Actuation Logic and Actuation Outputs for the following functions:

- Main Steam Line Isolation,
- Main Feedwater Isolation, ~~and~~
- Emergency Feedwater Actuation~~.~~, and
- Block Turbine Bypass and Cooldown Valves.

Condition T applies when the Required Action and associated Completion Time for Condition J or S have not been met. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours (12 hours total time). 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The ~~ESFAS~~PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 6 and Reference 7. The ACTUATION LOGIC TEST is a check of the ESFAS software memory integrity to ensure there is no change to the internal ESFAS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 6 and Reference 7. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, and the 24 month CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the 24 month COT, the 24 month ACTUATION LOGIC TEST and the 24 month ESFAS and SLS TADOT for the non-digital side of the output module. The Channel CALIBRATION, COT, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

The ACTUATION LOGIC TEST interval of 24 months with the self test capability is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a COT.

The PSMS is self-tested on an automatic basis from the digital side of all input modules to the digital side of all output modules. Self-testing encompasses all Trip Setpoints and trip functions. The self-testing is described in Reference 6 and Reference 7. ESFAS setpoint and bistable functions are implemented within the RPS. Therefore, the COT is a check of the RPS software memory integrity to ensure there is no change to the internal RPS software that would impact its functional operation, including digital Trip Setpoint values or the continuous self-test function. The software memory integrity test is described in Reference 6 and Reference 7.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A COT ensures the entire channel will perform the intended Function.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the 24 month CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the 24 month COT and the 24 month TADOT for the non-digital side of the output module. The Channel CALIBRATION, COT and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

The COT interval of 24 months with the self test capability is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a TADOT for the Actuation Outputs of all ESFAS functions. This function actuates the outputs of the SLS.

Therefore, this test is typically conducted in conjunction with testing the plant process components. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a TADOT for the Loss of Offsite Power, Function. The LOP inputs to the ESFAS are tested up to, and including, the signal status readout on a digital display.

The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.8

This SR ensures the response times for all ESFAS functions are less than or equal to the maximum values assumed in the accident analysis. Accident analysis response time values are defined in Reference 2. Individual component response times are not modeled in the analyses.

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in ~~both~~all trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

Electro-mechanical components in the ESFAS have aging or wear-out mechanisms that can impact response time. Response time for other components may be affected by random failures or calibration discrepancies, which are detectable by other testing and calibration methods required by other surveillances.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor, signal conditioning and actuation logic response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications.

The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that clarifies that the tests for the turbine driven EFW pumps ~~is tested~~are conducted within 24 hours after reaching 1000 psig in the SGs.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a TADOT for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB cycle. Each RTB status contact is tested up to, and including, the signal status readout on a digital display. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. NUREG-0737, “Clarification of TMI Action Plan Requirements.”
2. FSAR ~~Chapter 7~~Section 7.3.1.
3. FSAR Chapter 15.
4. IEEE-603-1991.
5. 10 CFR 50.49.
6. MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), “Safety I&C System Description and Design Process.”
7. MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), “Safety System Digital Platform – MELTAC.”
8. FSAR ~~Chapter 8~~Section 8.3.1.
9. 10 CFR 50.36.
~~10. Chapter 6~~.

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The purpose of displaying PAM parameters is to assist MCR personnel in evaluating the safety status of the plant. PAM parameters are direct measurements or derived variables representative of the safety status of the plant. The primary function of the PAM parameters is to aid the operator in the rapid detection of abnormal operating conditions. As an operator aid, the PAM variables represent a minimum set of plant parameters from which the plant safety status can be assessed.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by FSAR Chapter 7 (Ref. 4) addressing the recommendations of Regulatory Guide 1.97 (Ref. 1) as required by Supplement 1 to NUREG-0737 (Ref. 2).

The instrument channels required to be OPERABLE by this LCO include parameters based on IEEE 497-2002 (Ref. 5), which is endorsed by Regulatory Guide 1.97 (Ref. 1), identified as Type A, B and C variables.

Type A, B, and C variables are the key variables deemed risk significant because they are needed to:

Type A

Take planned manually controlled actions for accomplishment of safety-related functions for which there is no automatic control.

Type B

Assess the process of accomplishing or maintaining plant critical safety functions.

Type C

Indicate potential for a breach of fission product barriers.

Indicate an actual breach of fission product barriers.

BASES

BACKGROUND (continued)

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the operability of Type A, B and C variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of PAs), e.g., loss of coolant accident (LOCA),
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function,
- Determine whether systems important to safety are performing their intended functions,
- Determine the likelihood of a gross breach of the barriers to radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Type A variables, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses instruments that have been designated Type B and C.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

BASES

LCO (continued)

SG Water Level (Wide Range) is used to:

- identify the faulted SG following a tube rupture,
- verify that the intact SGs are an adequate heat sink for the reactor,
- determine the nature of the accident in progress (e.g., verify an SGTR), and
- verify unit conditions for termination of SI during secondary unit HELBs outside containment.

Operator action is based on the control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. Extended startup range level is a Type A variable because the operator must manually raise and control SG level to establish boiler condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the boiler condenser setpoint. This function is an alternate mean with EFW Flow.

12,13,14,15. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

Twenty six core exit thermocouples are provided for measuring core cooling as the post accident monitors. These thermocouples are arranged in two safety trains and a train consists of thirteen thermocouples. These thermocouples in each train are distributed at the exit of the core nearly uniformly and a minimum of 2 thermocouples are provided for each core quadrant. These distributed thermocouples provide adequate information of temperature distribution of core exit fluid. The uniform distributions of two train thermocouples ensure the adequate information of radial temperature distribution in a single failure condition.

16. Emergency Feedwater Flow

EFW Flow is provided to monitor operation of decay heat removal via the SGs.

~~Redundant monitoring capability is provided by two independent trains of instrumentation for each SG.~~

BASES

LCO (continued)

EFW flow is used three ways:

- to verify delivery of EFW flow to the SGs,
- to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range), and
- to regulate EFW flow so that the SG tubes remain covered.

This function is an alternate mean with SG Water Level.

17. Degrees of Subcooling

The Degrees of Subcooling is provided for verification of core cooling. Degrees of Subcooling utilizes sensors for RCS cold and hot leg temperatures, core exit temperature and RCS pressure. The saturation temperature is calculated from minimum temperature input. The temperature subcooled or superheated margin is the difference between the saturation temperature and the sensor temperature input. Two temperatures subcooled or superheated margin presentation are available as follows:

- RCS saturation margin – the temperature saturation margin based on the difference between the saturation temperature and the maximum temperature from the RTDs in the hot and cold legs.
- Upper head saturation margin – temperature saturation margin based on the difference between the saturation temperature and the core exit temperature.

18. Main Steam Line Pressure

Steam Generator Pressure is provided to monitor operation of decay heat removal via the SGs.

19. Emergency Feedwater Pit Level

EFW Pit Level is provided to ensure water supply for emergency feedwater (EFW). The EFW Pits provide the ensured safety grade water supply for the EFW System. There are two identical EFW Pits, each of which supplies one motor driven and one turbine driven EFW pump. Redundant level indication for each EFW Pit is displayed in the main control room.

BASES

ACTIONS (Continued)

Required Action C.2 is modified by a Note that indicates C.2 is only required to be performed when the Emergency Feedwater Pit Level is inoperable.

D.1

Condition D applies when the Required Action and associated Completion Time of Condition C is not met. Required Action D.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C is not met and Table 3.3.3-1 directs entry into Condition E, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

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

F.1

At this unit, alternate means of monitoring Reactor Vessel Water Level and Containment High Area Radiation have been developed and tested. Also, alternate means of the RCS Hot Leg Temperature (Wide Range) and RCS Cold Leg Temperature (Wide Range) have been developed and tested. Also, alternate means of Steam Generator Water Level (Wide Range) and Emergency Feedwater Flow have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.5, in the Administrative Control section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels,

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

BASES

SR 3.3.3.2

CHANNEL CALIBRATION is a complete check of the instrument loop, from sensor to VDU as described in Reference 3. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.97, Rev. 4.

2. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

3. MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), “Safety I&C System Description and Design Process.”

4. FSAR ~~Chapter 7~~Section 7.5.

5. IEEE 497-2002.

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown Console (RSC)

BASES

BACKGROUND

The RSC provides sufficient displays and controls for the control room operator to place and maintain the unit in a safe shutdown condition (MODE 3) from a location outside the Main Control Room if the control room becomes inaccessible.

With the unit in MODE 3, the Emergency Feedwater (EFW) System and the steam generator (SG) safety valves or the main steam depressurization valves (MSDVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the EFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control at the RSC, and place and maintain the unit in MODE 3 for an extended period of time.

APPLICABLE SAFETY ANALYSES

The RSC is located outside the control room with a capability to promptly shutdown, cooldown and maintain the unit in a safe condition in MODE 3 (Ref. 4).

The criteria governing the design and specific system requirements for remote shutdown~~of the Remote Shutdown System~~ are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1). These criteria are applied to the RSC of the US-APWR.

The RSC satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 2).

The RSC LCO provides the OPERABILITY requirements for the RSC, which includes the displays and controls necessary to place and maintain the unit in MODE 3 and the ability to transfer control from the MCR to the RSC.

LCO

Display and Control

The displays and controls at the RSC are functionally the same as the displays and controls used by the operator to achieve and maintain MODE 3 from the main control room. These displays and controls are provided by four trains of Safety VDUs, and non-safety Operational VDUs. MODE 3 can be achieved and maintained using only safety related plant equipment which is controlled and monitored from Safety VDUs or Operational VDUs.

BASES

LCO (continued)

Non-safety plant equipment is controlled and monitored from the Operational VDUs at the RSC. This equipment is provided for convenience and is not necessary to achieve or maintain MODE 3. Therefore the Operational VDUs are not covered by this LCO.

Transfer of Control

The controls in the MCR are normally enabled, while the controls at the RSC are normally disabled. Actuation of Transfer Switches disables the controls in the MCR and enables the controls at the RSC. There are two Transfer Switches for each safety train of the PSMS and two transfer switches for the PCMS. Activating both transfer switches for a train, transfers the controls for that train.

The RSC equipment covered by this LCO does not need to be continuously energized to be considered OPERABLE. However, it is necessary to energize this equipment for surveillance testing.

APPLICABILITY

The RSC LCO is applicable in MODES 1, 2 and 3. This applicability recognizes the need for being able to place and maintain the unit in a safe shutdown condition from a location outside the main control room if the MCR becomes inaccessible while the RCS contains a large amount of energy.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

A.1

Condition A addresses the situation where the Remote Shutdown Console is inoperable. This includes the Display and Control Function and the Transfer of Control Function.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that all logic and communications with the PSMS for the Transfer of Control Function is OPERABLE. It also verifies that all logic functions within the PSMS associated with controls and indications at the RSC are OPERABLE.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.3

SR 3.3.4.3 is the performance of a Safety VDU test for all Safety VDUs on the RSC. The Safety VDU Test is explained in Reference 3.

This Surveillance confirms the Safety VDU is ~~capability~~capable of providing all display and control functions for the RSC. This test overlaps with the Actuation Logic Test of SR 3.3.4.2 to ensure the Display and Control Function is OPERABLE.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. 10 CFR 50.36.

3. MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), "Safety I&C System Description and Design Process."

4. FSAR ~~Chapter 7~~Section 7.4.1.

5. MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), "Safety System Digital Platform – MELTAC."

B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Class 1E Gas Turbine Generator (GTG) Start Instrumentation

BASES

BACKGROUND

The Class 1E GTG provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs in the switchyard. There are four LOP start signals, one for each 6.9 kV Class 1E bus.

Three undervoltage relays with inverse time characteristics are provided on each 6.9 kV Class 1E bus for detecting a sustained degraded voltage condition or a loss of bus voltage. The relays are combined in a two-out-of-three logic to generate an LOP signal ~~if~~when the voltage is ~~below 70%~~dropped before reaching the loss of voltage limit for a short time or ~~below 90%~~before reaching the degraded voltage limit for a long time. The LOP start actuation is described in Reference 1.

The Allowable Value in conjunction with the Trip Setpoint and LCO establishes the threshold for Engineered Safety Features Actuation System (ESFAS) action to prevent exceeding acceptable limits such that the consequences of Postulated Accidents (PAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the setpoint is found not to exceed the Allowable Value during the CHANNEL CALIBRATION. Note that although a channel is OPERABLE under these circumstances, the setpoint must be left adjusted to within the established calibration tolerance band of the setpoint in accordance with uncertainty assumptions stated in the referenced setpoint methodology, (as-left-criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

Allowable Values and LOP Class 1E GTG Start Instrumentation Setpoints

Setpoints adjusted consistent with the requirements of the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed. The time delay of the Class 1E GTG starting initiated by LOOP signal is considered as mitigation system time delay in the analysis presented in Chapter 15.

Allowable Values and/or Nominal Trip Setpoints are specified for each Function in SR 3.3.5.3. The trip setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered

BASES

BACKGROUND (Continued)

OPERABLE. Operation with a trip setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation.

APPLICABLE SAFETY ANALYSES ~~System (ESFAS)~~

The LOP Class 1E GTG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS) Accident analyses credit the loading of the Class 1E GTG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual Class 1E GTG start has historically been associated with the ESFAS actuation. The Class 1E GTG loading has been included in the delay time associated with each safety system component requiring Class 1E GTG supplied power following a loss of offsite power. The analyses assume a non-mechanistic Class 1E GTG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

The required channels of LOP Class 1E GTG start instrumentation, in conjunction with the ESF systems powered from the Class 1E GTGs, provide unit protection in the event of any of the analyzed accidents discussed in ~~Reference 2~~FSAR Chapter 15, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 100 second Class 1E GTG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate Class 1E GTG loading and sequencing delay.

The LOP Class 1E GTG start instrumentation channels satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 4).

LCO

The LCO for LOP Class 1E GTG start instrumentation requires that three channels per bus of both the loss of voltage and degraded voltage Functions shall be OPERABLE in MODES 1, 2, 3, and 4, as well as whenever the associated GTG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown." A Class 1E GTG is not required to be OPERABLE if its associated Class 1E 6.9 kV bus is not powering any required ESF loads. Therefore the associated Class 1E 6.9 kV bus is not required.

Loss of the LOP Class 1E GTG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite

BASES

ACTIONS (Continued)

B.1

Condition B applies when ~~more than one~~two or more loss of voltage or ~~more than one~~two or more degraded voltage channel per required Class 1E 6.9 kV bus are inoperable.

Required Action B.1 requires restoring all but one channel per required Class 1E 6.9 kV bus to OPERABLE status. The 1 hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring an LOP start occurring during this interval.

C.1

Condition C applies to each of the LOP Class 1E GTG start Functions when the Required Action and associated Completion Time for Condition A or B are not met.

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the Class 1E GTG made inoperable by failure of the LOP Class 1E GTG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined based on a combination of the channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

SURVEILLANCE REQUIREMENTS (continued)

A CHANNEL CHECK may be conducted manually or automatically. For the US-APWR an automated CHANNEL CHECK is normally conducted continuously. Where the CHANNEL CHECK is conducted automatically, an alarm shall be generated when the agreement criteria is not met.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.2

SR 3.3.5.2 is the performance of a TADOT for the LOP undervoltage relays and their interface to the PSMS. For these tests, the undervoltage relay is confirmed to actuate ~~for gross loss of voltage conditions~~with reasonable proximity to the Nominal Trip Setpoints. Undervoltage trip setpoints Allowable Values and time delays are verified during CHANNEL CALIBRATION, SR 3.3.5.3.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

CHANNEL CALIBRATION for a binary process measurement is a complete check of the instrument loop, including the sensor and interface to the PSMS, as described in Reference 2. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

In SR 3.3.5.3, the values specified for Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

In SR 3.3.5.3, the values specified for Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.4

SR 3.3.5.4 is the performance of an ACTUATION LOGIC TEST. The Class 1E GTG start logic within the PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 2 and 3. The ACTUATION LOGIC TEST is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 2 and 3. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the 24 month CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the 24 month ACTUATION LOGIC TEST, and the 24 month ESFAS and SLS TADOT for the non-digital side of the output module. The Channel CALIBRATION, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

The ACTUATION LOGIC TEST interval of 24 months with the self test capability is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.5

SR 3.3.5.5 is the performance of a TADOT for the Actuation Outputs to start the Class 1E GTGs. This function actuates the outputs of the SLS. Therefore, this test is typically conducted in conjunction with testing the Class 1E GTG. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Actuation Outputs are solid state devices. Since this test is conducted in conjunction with testing for the Class 1E GTG, this test may be conducted more frequently, as may be required for the Class 1E GTG.

REFERENCES

1. FSAR ~~Chapter 8~~Section 8.3.1.
2. MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), “Safety I&C System Description and Design Process.”
3. MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), “Safety System Digital Platform – MELTAC.”
4. 10 CFR 50.36.

BACKGROUND (continued)

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

DAAC Actuation Logic and Actuation Outputs

The DAAC provides the decision logic processing of outputs from the signal processing equipment bistables. To prevent spurious actuation, two subsystems of DAAC, each performing the same functions, are provided in a two out of two configuration. The subsystems are designed such that testing may be accomplished while the reactor is at power and without causing trip/actuation. If one subsystem is actuated for maintenance or test purposes, there will be no reactor trip, turbine trip or ESF actuation for the unit. If both subsystems are actuated, a reactor trip, turbine trip and/or ESF actuation will result. Each subsystem is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to not trip/actuate in the event of a loss of power, to prevent spurious actuation.

The DAAC performs the decision logic for actuating a reactor trip, turbine trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

Within each DAAC subsystem, the bistable outputs from the signal processing equipment are combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will send actuation signals, to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition, via PSMS Power Interface modules, if necessary. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases. Output signals from each ~~DAAC channel~~DAAC Subsystem are combined in a two-out-of-two logic within Rod Drive Motor-Generator set trip ~~devise~~devices or the Power Interface module for each plant component. When each DAAC subsystem is tested, the interface to the Power Interface is tested. When plant components are actuated from the PSMS, either for testing or control, the PSMS output signals overlap with the DAAC output signals within the Power Interface. This overlap completes the DAS Function testing. Testing of PSMS components is per LCO 3.3.2.

BASES

BACKGROUND (continued)

Rod Drive Motor-Generator sets

The Rod Drive Motor-Generator sets are the electrical power supply for the CRDMs. Tripping the Rod Drive Motor-Generator sets trip ~~devise~~devices interrupts power to the CRDMs, which allows the control rod shutdown banks and control banks to fall into the core by gravity. There are two Rod Drive Motor-Generator sets operating in parallel. The DAS trips both Rod Drive Motor-Generator sets trip ~~devise~~devices.

The DAS interface to the Rod Drive Motor-Generator sets is via hardwired circuit. This interface may be tested, with no reactor trip, as described ~~above~~in subsection 7.8.2.4. Actual tripping of the Rod Drive Motor-Generator set may be tested from the DAS. Rod Drive Motor-Generator sets may be tripped one at a time for testing.

Diverse Human System Interface Panel (DHP)

The DHP provides Manual Initiation switches for all DAS automatic actuation functions and for additional functions that are required, per the D3 Coping Analysis, to control all critical safety functions. Manual Initiation switches are not redundant. To prevent spurious actuation due to a failure of any of the above switches, a separate manual actuation permissive switch is provided. This is referred to as the “Permissive Switch for DAS HSI.”

The DHP also provides indications, per the D3 Coping Analysis, to monitor all critical safety functions.

The DHP also provides indications, per the D3 Coping Analysis, to monitor RCS Leakage.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The DAS is required to provide a diverse capability to trip the reactor and actuate the specified safety-related equipment. The DAS is not credited for mitigating accidents in the FSAR Chapter 15 safety analyses. The DAS satisfy Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 5).

The DAS LCO provides the requirements for the OPERABILITY of the DAS necessary to place the reactor in a shutdown condition and to remove decay heat in the event that required PSMS components do not function due to CCF.

A channel is OPERABLE provided the "as-found" accuracy value does not exceed its associated Allowable Value. A trip setpoint may be set more conservative than the Trip Setpoint as necessary in response to plant conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The DAS is required to be OPERABLE in the MODES specified in Table 3.3.6-1. All function of ~~T~~the DAS ~~Reactor Trip Function is~~are required to be OPERABLE in MODES 1, 2 and 3 with the pressurizer pressure > P-11.

In Table 3.3.6-1, the values specified for Allowable Values and Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

DAS functions are as follows:

1. Reactor Trip, Turbine Trip and Main Feedwater Isolation

   a. Manual Initiation

      The LCO requires 1 channel to be OPERABLE. This consists of the Reactor Trip, Turbine Trip and Main Feedwater Isolation - Manual Initiation switch. This function requires operation of the Permissive Switch for DAS HSI. The Permissive Switch for DAS HSI is common for all DAS Manual Initiation/Control Functions. The operator can initiate this function at any time by operation of both of these switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   b. Automatic Actuation Logic and Actuation Outputs

      This LCO requires two channels to be OPERABLE. Actuation logic consists of all circuitry housed within the DAAC, up to the Power Interface modules responsible for actuating the ESF equipment.

   c. Low Pressurizer Pressure

      There are four Low Pressurizer Pressure channels in two-out-of-four voting logic. This automatic function is automatically blocked when status signals (P-4) are received indicating that the minimum combination of the RTBs have actuated for the RT function. The LCO requires 2 Low Pressurizer Pressure channels to be OPERABLE.

   d. High Pressurizer Pressure

      There are four High Pressurizer Pressure channels in two-out-of-four voting logic. This automatic function is

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

automatically blocked when status signals (P-4) are received indicating that the minimum combination (2-out-of-4) of the RTBs have actuated for the RT function. The LCO requires 2 High Pressurizer Pressure channels to be OPERABLE.

e. Low Steam Generator Water Level

There is one Low SG Water Level channel for each SG (four total). The LCO requires 1 Low SG Water Level channel to be OPERABLE on any 2 Steam Generators. These signals from each SG are processed through two-out-of-four voting logic. The D3 Coping Analysis demonstrates that the two-out-of-four voting logic is adequate for all secondary events including loss of feedwater and SG rupture. This automatic function is automatically blocked when status signals (P-4) are received indicating that the minimum combination (2-out-of-4) of the RTBs have actuated for the RT function.

f. Rod Drive Motor-Generator set

This LCO requires two channels, one for each Motor-Generator set, to be OPERABLE. Each channel trips one Motor-Generator set. Both Motor-Generator sets must be tripped for this Reactor Trip Function.

2. Emergency Feedwater Actuation

a. Manual Initiation

Manual Initiation consists of the same features and operates in the same manner as described for DAS Function 1.a.

b. Automatic Actuation Logic and Actuation Outputs

Automatic actuation logic and actuation outputs consist of the same features and operate in the same manner as described for DAS Function 1.b.

c. Low Steam Generator Water Level

The Low Steam Generator Water Level channels consist of the same features and operate in the same manner as described for DAS Function 1.e. This automatic function is automatically blocked when status signals are received

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

indicating that the ESFAS EFW function has actuated correctly. Correct actuation is indicated when 2-out-of-4 status signals are received from limit switch contacts on the steam inlet valves to the turbine driven EFW pumps and from auxiliary contacts on the motor starters controlling the motor driven EFW pumps.

3. ECCS Actuation

a. Manual Initiation

Manual Initiation consists of the same features and operates in the same manner as described for DAS Function 1.a.

4. Containment Isolation

a. Manual Initiation

Manual Initiation consists of the same features and operates in the same manner as described for DAS Function 1.a.

5. EFW Isolation Valves

a. Manual Control

There are separate EFW Isolation Valves Control switches for each Steam Generator. The LCO requires 1 channel to be OPERABLE for each of four Steam Generators. This consists of the EFW Isolation Valves - Manual Control switch, and the Permissive Switch for DAS HSI. The Permissive Switch for DAS HSI is common for all DAS Manual Initiation/Control Functions. The operator can initiate this function at any time by operation of both of these switches in the control room.

6. Pressurizer Safety Depressurization Valves

a. Manual ~~Initiation~~Control

Manual Initiation consists of the same features and operates in the same manner as described for DAS Function 1.a.

BASES

SURVEILLANCE REQUIREMENTS (continued)

REFERENCES

1. MUAP-07006-P (Proprietary) and MUAP-07006-NP (Non-Proprietary), “Defense-in-Depth and Diversity.”

2. MUAP-07014-P (Proprietary) and MUAP-07014-NP (Non-Proprietary), “Defense-in-Depth and Diversity Coping Analysis.”

3. FSAR ~~Chapter 7~~Section 7.8.

4. 10 CFR 50.49.

5. 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure limit and RCS average temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 551°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical (k_{eff} ≥ 1.0) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with k_{eff} ≥ 1.0, LCO 3.4.2 is applicable since the reactor can only be critical (k_{eff} ≥ 1.0) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with K_{eff} < 1.0 in an orderly manner and without challenging plant systems.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

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

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

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

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an~~An~~ adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

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. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 5).

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference ~~1~~7 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. Restoration of P/T parameters to the analyzed range reduces the RCPB stress.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

REFERENCES (continued)

3. ASTM E 185-82, July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. FSAR Subsection 5.3.2.1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the four pump coastdown, single pump locked rotor, single pump, broken shaft, or coastdown, and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the four RCS loop operation. For four RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 120% RTP. This is the design overpower condition for four RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (CS/RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

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

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or ~~one~~two RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that ~~two~~additional paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS 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 ~~be~~are OPERABLE in MODE 4 and that one of ~~these loops be~~the RCS loops or two of the RHR loops are in operation. ~~The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops.~~ Any one RCS loop or two RHR loops in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or CS/RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of ~~either~~ RCS or two loops of RHR provides sufficient circulation for these purposes. However, ~~two loops consisting of any combination~~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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" MODE 6), and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

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

A.2

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

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if ~~a~~two RHR loops ~~is~~are OPERABLE. With no RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a RHR loop, rather than a cooldown of extended duration.

ACTIONS (continued)

B.1 and B.2

If two or more required loops are inoperable or a required loop(s) ~~is~~are not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until ~~one loop is~~the required loop(s) are restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification that the required RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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 ≥ 13%. If the SG secondary side narrow range water level is < 13%, 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND	In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (CS/RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of four RHR loops connected to the RCS, each loop containing a CS/RHR heat exchanger, a CS/RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. Two CS/RHR pumps circulate the water through the RCS at a sufficient rate to prevent boric acid stratification. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least two RHR loops for decay heat removal and transport. The flow provided by two RHR loops is adequate for decay heat removal. The other intent of this LCO is to require that a third path be available to provide redundancy for heat removal. The LCO provides for redundant paths of decay heat removal capability. The first two paths can be RHR loops that must be OPERABLE and in operation. The third path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ 13% to provide an alternate method for decay heat removal via natural circulation (Ref. 1).
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

LCO (continued)

Note ~~2~~3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

CS/RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. Two loops~~3~~ of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 13%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"

LCO 3.4.5, "RCS Loops - MODE 3;"

LCO 3.4.6, "RCS Loops - MODE 4;"

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation -High Water Level" (MODE 6)," and

LCO 3.9.6,"Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6)."

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 (CS/RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. In MODE 5 with loops not filled, only CS/RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least two CS/RHR pumps for decay heat removal and transport and to require that three 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 two RHR loops is adequate for heat removal and for boron mixing. RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	The purpose of this LCO is to require that at least three RHR loops be OPERABLE and two 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 two running CS/RHR pumps meets the LCO requirement for two loops in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations. Note 1 permits one CS/RHR pump to be removed from operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping one CS/RHR pump is to be limited to situations when the outage time is short and core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1 is maintained or draining operations when RHR forced flow is stopped.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and ~~pressurizer power operated relief valves~~safety depressurization valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "~~Pressurizer Power Operated Relief Valves (PORVs)~~Safety Depressurization Valves (SDVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

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

Safety analyses does not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure (Ref. 1).

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

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 2668 cubic feet, which is equivalent to 92%, 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.

The LCO requires three groups of OPERABLE pressurizer heaters, each with a capacity ≥ 120 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of 120 kW is derived from the use of three heaters rated at 46.8 kW each. The amount needed to maintain pressure is dependent on the heat losses.

ACTIONS (continued)

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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 pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 15~~Subsection 15.0.0.2.2.

2. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), ~~2735~~2733.5 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The capacity for each valve, 432,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE SAFETY ANALYSES	All accident and safety analyses that require safety valve actuation assume operation of four pressurizer safety valves to limit increases in RCS pressure. Accidents that could result in overpressurization if not properly terminated include: a. Loss of external electrical load, b. Loss of normal feedwater flow, c. Reactor coolant pump shaft break, d. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition, and e. Spectrum of rod cluster control assembly ejection accidents Detailed analyses of the above transients are contained in FSAR Chapter 15 (Ref. 2). Safety valve actuation is required in events a, b, and d (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions (Ref. 2 and 3). Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	The four pressurizer safety valves are set to open at the RCS design pressure 2485 psig, and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.
APPLICABILITY	In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of four valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± ~~3~~1% for OPERABILITY; ~~however,~~and the valves are reset to remain within ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, NB 7500.

2. FSAR Chapter 15.

3. FSAR ~~Chapter 5~~Subsection 5.2.2.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES

For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The SDVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

Pressurizer SDVs satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The LCO requires the SDVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR and to depressurize RCS associated with safety shutdown.

By maintaining two SDVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. The block valves may be OPERABLE when closed to isolate the flow path of an inoperable SDV that is capable of being manually cycled (e.g., as in the case of excessive SDV leakage). Similarly, isolation of an OPERABLE SDV does not render that SDV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE SDV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

BASES

APPLICABILITY

The SDVs are required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event and to depressurize the RCS during safety shutdown.

The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant.

ACTIONS

Note 1 has been added to clarify that all pressurizer SDVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

SDVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, either the SDVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the SDVs to eliminate the problem condition.

Quick access to the SDV for pressure control can be made when power remains on the closed block valve and furthermore pressurizer safety valve can be expected to open at its set pressure. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, B.3.1, and B.3.2

If one ~~PORV~~SDV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the SDVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one SDV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable SDV to OPERABLE status. If the SDV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

Required Action B.3.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

BASES

ACTIONS (continued)

F.1~~, F.2, and F.3~~

If more than one block valve is inoperable, it is necessary to restore at least one block valves within 2 hours. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each ~~PORV~~SDV. Operating a SDV through one complete cycle ensures that the SDV can be manually actuated for mitigation of an SGTR. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.32, February 1977.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that t~~T~~he reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding LTOP arming temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. When the RHR system is placed in service, overpressure prevention is provided by two RHR suction relief valves or by a depressurized RCS and a sufficient sized RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the RHR suction relief valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RHR suction relief valve or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters,

b. Loss of RHR cooling, or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but two safety injection pumps and one charging pump incapable of injection,

b. Deactivating the accumulator discharge isolation valves in their closed positions, and

APPLICABLE SAFETY ANALYSES (continued)

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either two RHR suction relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when two safety injection pumps and one charging pump are actuated. Thus, the LCO allows only two safety injection pumps and one charging pump OPERABLE during the LTOP MODES.

Since neither two RHR suction relief valves nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (195°F and below) than that of the LCO (LTOP arming temperature and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at LTOP arming temperature specified in the PTLR.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of two SI pumps and one charging pump OPERABLE and SI actuation enabled.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints. Analyses must show that two RHR suction relief valves lifting at its specified setpoint will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, ~~an~~two RHR suction relief valves will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of ~~2.6~~4.7 square inches is capable of mitigating the allowed LTOP overpressure transient.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capacity of a vent this size is greater than the flow possible from either the mass or heat input transient, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The required vent area may be created by removing SDVs with their associate block valves secured in the open position or by removal of pressurizer safety valves.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient. Since neither two RHR suction relief valves nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

To limit the coolant input capability, the LCO requires that a maximum of two safety injection pumps and one charging pump be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The LCO is modified by two Notes. Note 1 allows two charging pumps to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

LCO (continued)

a. Two OPERABLE RHR suction relief valves, or

An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ ~~2.6~~4.7 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ LTOP arming temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above LTOP arming temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 when RHR is isolated or RCS temperature is above LTOP arming temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure with little or no time allowed for operator action to mitigate the event.

ACTIONS

A NOTE prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and B.1

With three or more safety injection pumps, or two or more charging pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

BASES

ACTIONS (continued)

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to >LTOP arming temperature specified in the PTLR, an accumulator pressure of 695 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1 and E.2

With the RCS pressurized and an RHR suction relief valve inoperable, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The RHR suction relief valve is considered inoperable if one or both of the RHR suction isolation valves are closed such that the RHR suction relief valve in that RHR train cannot perform its intended safety function, or if the valve itself will not operate to perform its intended safety function.

Under these conditions, Required Action E.1 or E.2 provides two options, either of which must be accomplished in 12 hours. If the RHR suction relief valve cannot be restored to OPERABLE status, the RCS must be depressurized and vented with an RCS vent which provides a flow area sufficient to mitigate any of the design low temperature overpressure events.

The 12 hour completion time represents a reasonable time to repair the relief valve, open the RHR isolation valves or otherwise restore the system to OPERABLE status, or depressurize and vent the RCS, without imposing a lengthy period when the LTOP system is not able to mitigate a low temperature overpressure event.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of two safety injection pumps and a maximum of one charging pump are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out.

The safety injection pumps and charging pump are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.4

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying both RHR suction isolation valves are open. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified to be opened. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.5

The RCS vent of ≥ ~~2.6~~4.7 square inches is proven OPERABLE by verifying its open condition either:

a. A valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or

b. Other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.6

The RHR suction relief valves shall be demonstrated OPERABLE by verifying that both RHR suction isolation valves in one flow path are open and by testing ~~it~~them in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.~~2~~4 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO. The ASME Code (Ref. ~~4~~5), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.7

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open. This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

The RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The “locked open” valve must be locally verified in its open position with the manual actuator locked in its inactive position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix G.

2. Generic Letter 88-11.

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

4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~FSAR Subsection 5.2.2.

5. ASME, Code for Operation and Maintenance of Nuclear Power Plants.

6. 10 CFR 50, Section 50.46.

7. 10 CFR 50, Appendix K.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 600 gallons per day. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is equivalent to the conditions assumed in the safety analysis (Ref. 3).

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.

b. Unidentified LEAKAGE

~~One~~0.5 gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for containment spray.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System,

BACKGROUND (continued)

b. Safety Injection System, and

c. Chemical and Volume Control System.

The PIVs are listed in FSAR Chapter 3. (Ref. 6).

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 900 psig, and 900 psig design is able to bear the RCS pressure without pipe rupture, overpressurization failure of the RHR low pressure line is prevented, thus preventing a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

BASES

ACTIONS (continued)

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If leakage cannot be reduced, the system cannot be isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

The inoperability of the RHR suction valve interlock renders the RHR suction isolation valves incapable of preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR suction valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the suction valve interlock function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2

Verifying that the RHR suction valve interlock is OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond its design pressure of900 psig. The interlock setpoint is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. FSAR ~~Chapter 3~~Subsection 3.9.6.3.4.
7. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE and air cooler condensate flow rate monitor are instrumented to alarm for increases of greater than or equal to ~~1.0~~0.5 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivity~~ies~~ of 10^{-9} μCi/cc radioactivity for particulate monitoring ~~and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are~~is practical for these leakage detection systems. Radioactivity detection system~~s are~~is included for monitoring ~~both~~ particulate ~~and gaseous activities~~activity because of their sensitivity~~ies~~ and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

BASES

BACKGROUND (continued)

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from air coolers. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR Chapter 5 (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

BASES

ACTIONS (continued)

B.1.1, B.1.2, B.2.1, and B.2.2

With both ~~gaseous and~~ particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances performed every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

With the required containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum ~~total effective~~ dose ~~equivalent~~ that an individual at the ~~site~~exclusion area boundary can receive for 2 hours ~~during~~following an accident, or at low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50. 34 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are ~~held to a small fraction of the 10 CFR 50.34 limits~~appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of ~~iodines and noble gases~~radionuclides in the reactor coolant. The LCO limits are established to minimize the ~~offsite radioactivity~~ dose consequences in the event of a steam system piping failure or a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 ~~specific activity~~. ~~The LCO limits are established by assuming 1 % failed fuel.~~The allowable levels are intended to ensure that offsite and control 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 ensures that the resulting offsite and control room doses meet the appropriate SRP~~2 hour doses at the site boundary will not exceed a small~~ acceptance criteria following a steam system piping failure or ~~fraction of the 10 CFR 50.34 dose guideline limits following~~ a SGTR accident. The ~~SGTR~~ safety ~~analysis~~analyses (Ref. ~~2~~3 and 4) assume~~s~~ the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 600 gpd exists. The safety ~~analysis~~analyses assume~~s~~ the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity."

The analys~~i~~es for the steam system piping failure and SGTR accidents establish~~es~~ the acceptance limits for RCS specific activity. Reference to ~~this~~these ~~analysis~~analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety ~~analysis~~analyses ~~is for~~consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the ~~I-131 activity in the reactor coolant~~rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after ~~the accident~~a steam system piping failure (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity at 60 μCi/gm DOSE EQUIVALENT I-131 due to ~~a pre-accident~~an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity ~~in the reactor coolant~~ is assumed to be 300 μCi/gm DOSE EQUIVALENT XE-133. ~~These limits are established by assuming 1% failed fuel.~~

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. If the reactor trip system has not automatically tripped the reactor, operators are assumed to manually trip the reactor.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the main steam relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The steam system piping failure radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The ~~specific~~ iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 300 μCi/gm DOSE EQUIVALENT XE-133 ~~specific activity in the reactor coolant is limited to 300 μCi/gm~~. The~~se~~ limits on specific activity ensure that offsite and control room~~the~~ doses ~~to an individual at the site boundary during a Design Basis Accident (DBA)~~ will meet the appropriate SRP acceptance criteria (Ref. 2)~~be a small fraction of the limits specified in 10 CFR 50.34~~.

The steam system piping failure and SGTR accident ~~analysis~~analyses (Refs. ~~2~~3 and 4) show~~s~~ that the ~~offsite~~calculated doses ~~levels~~ are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a steam system piping failure or~~an~~ SGTR, lead to ~~site boundary~~ doses that exceed the SRP acceptance criteria (Ref. 2)~~10 CFR 50.34 dose guideline limits~~.

APPLICABILITY

In MODES 1 ~~and~~, 2, ~~and in MODE~~ 3 and 4 ~~with RCS average temperature ≥ 500°F~~, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 ~~specific activity are~~is necessary to ~~contain~~limit the potential consequences of ~~an~~a steam system piping failure or SGTR to within the SRP acceptance criteria (Ref. 2)~~acceptable site boundary dose values~~.

~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activiity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that ~~DOSE EQUIVALENT I-131~~the specific activity is ≤ 60 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is ~~done to~~ continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit~~s~~ within 48 hours. The Completion Time of 48 hours is ~~based on a reasonable time for normal iodine spikes to decay back to within the LCO limit. If the concentration cannot be restored to within the LCO limit within 48 hours, then the LCO violation did not result from normal iodine spiking~~acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a steam system piping failure 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), ~~while~~ relying on ~~the~~Required ~~ACTIONS~~Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 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.

B.1

With the DOSE EQUIVALENT XE-133 ~~specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.~~greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a steam system piping failure or SGTR occurring during this time period.

~~The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.~~

BASES

ACTIONS (continued)

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 Actions 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.

C.1

If ~~a~~the Required Action and the associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60 μCi/gm , the reactor must be brought to MODE 3 ~~with RCS average temperature < 500°F~~ within 6 hours and MODE 5 within 36 hours. The allowed Completion Time ~~of 6 hours is~~are reasonable, based on operating experience, to reach the required plant conditions ~~MODE 3 below 500°F~~ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a measure of the noble gas specific activity of the reactor coolant. This is a quantitative measure of radionuclides with half lives longer than 15 minutes., This Surveillance provides an indication of any increase in the release of noble gas activity from fuel clad defects.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with Tavg at least 500°F. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when increased releases of iodine from fuel defects are more apt to occur. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the SurveillanceFrequency Control Program. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and relief valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 600 gallons per day (gpd). For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.34 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase and core reflooding phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The safety analysis assumes values of 19,338 gallons and 19,734 gallons.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is ~~slightly~~3434 gallons larger than the deliverable volume for the accumulators, since the flow damper is near the bottom of the accumulators and the dead volume in each accumulator is 3434 gallons. For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The safety analysis treats the volume of water from the accumulator to the RCS isolation check valves as accumulator injection line.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

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 1920 psig, and the limits established in the SRs for contained volume, boron concentration, 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 SI pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1 and A.2

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

B.1 and B.2

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is reasonable based on the consideration that the conclusion of TSTF-370 is applicable to US-APWR. The basic design concept of US-APWR including the design base success criteria of accumulators, and the core damage scenarios after postulated LOCA events when an accumulator is inoperable are equivalent with conventional PWRs, and therefore, the TSTF-370 is considered applicable. Additionally, PRA studies in Chapter 19, Subsection 19.1.4.1.2 (Ref. 4) show low CDF increment under conditions where one accumulator is inoperable~~FSAR Chapter 19 (Ref. 4)~~. Required Action B.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage Pit (RWSP), because the water contained in the RWSP is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ 1920 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 1920 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

REFERENCES

1. FSAR ~~Chapter 6~~Subsection 6.2.1.

2. 10 CFR 50.46.

3. FSAR ~~Chapter 15~~Subsection 15.6.5.

4. FSAR ~~Chapter 19~~Subsection 19.1.4.1.2.

5. NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 Safety Injection System (SIS) - Operating

BASES

BACKGROUND

The function of the SIS 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 SIS operation: direct vessel injection (DVI) and hot leg recirculation. In the DVI phase, water is taken from the refueling water storage Pit (RWSP) and injected directly into the reactor vessel downcomer. After approximately 4 hours of DVI, the SIS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The SIS consists of four 50% capacity trains. The ECCS accumulators and the RWSP are also part of the ECCS, but are not considered part of an SIS flow path as described by this LCO.

The SIS flow paths consist of piping, valves~~,~~, and pumps such that water from the RWSP can be injected into the RCS following the accidents described in this LCO. The major component of each train is the SI pump. Each pump is capable of supplying 50% of the flow required to mitigate the accident consequences. Four 50% capacity SIS trains ensure 100% of the required flow is delivered to the reactor with one train out of service, while still meeting the single failure criterion.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

The SIS 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 SI pumps, as well as the maximum response time for their actuation. The SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the SI pumps. The SGTR and MSLB events also credit the SI 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 a single failure disabling one SI pump while another SI pump is assumed out of service, i.e., 2 SI pumps are assumed to operate (two Class 1E GTG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation) and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one SIS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the reactor vessel 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~~at least two SIS trains are available to ~~will~~ deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

Long term core cooling is achieved by using the SI pumps.

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

LCO

In MODES 1, 2, and 3, three independent (and redundant) SIS trains are required to ensure that sufficient SIS flow is available, assuming a single failure affecting ~~either train~~one of the three required trains. Additionally, individual components within the SIS trains may be called upon to mitigate the consequences of other transients and accidents.

BASES

APPLICABILITY (continued)

APPLICABILITY	In MODES 5 and 6, plant conditions are such that the probability of an event requiring SIS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1 and A.2

With one required train inoperable and at least 100% of the SIS flow equivalent to a two OPERABLE SIS train available, one inoperable train must be returned to OPERABLE status within 72 hours. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine the Risk Informed Completion Time (RICT). The 72 hour Completion Time refers ~~to~~is based on PRA analyses described in FSAR Chapter 19 (Ref. 5) and is a reasonable time for repair of many SIS components.

An SIS 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.

FSAR Chapter 19 (Ref. 5) has shown that the impact of having one required SIS train inoperable is sufficiently small to justify continued operation for 72 hours.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the SI pumps to the RCS is maintained. Misalignment of these valves could render ~~its associated~~two SIS trains inoperable. Securing these valves in position by removal of power ~~or by key locking the control in the correct position~~ ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.2

Verifying the correct alignment for manual and power operated valves in the SIS flow paths provides assurance that the proper flow paths will exist for SIS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

Periodic surveillance testing of SI pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

This Surveillance demonstrates that each SI pump starts on receipt of an actual or simulated ECCS actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.5

Periodic inspections of the ECC/CS STRAINER ensure that it is unrestricted and stays in proper operating condition. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR ~~Chapter 6~~Subsection 6.2.1.
4. FSAR ~~Chapter 15~~Subsection 15.6.5.
5. FSAR Chapter 19.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 Safety Injection System (SIS) - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "Safety Injection System (SIS) - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the SIS train requirement is two trains.

An SIS flow path consists of piping, valves, and an SI pump such that water from the refueling water storage pit (RWSP) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the SIS operational requirements are reduced. It is understood in these reductions that certain automatic ECCS actuation is not available. In this MODE, sufficient time exists for manual actuation of the required SIS to mitigate the consequences of a DBA.

Only two trains of SIS are required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The SIS trains satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

In MODE 4, two of the four independent (and redundant) SIS trains are required to be OPERABLE to ensure that sufficient SIS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWSP.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWSP to the RCS via the SI pumps to the reactor vessel direct injection nozzles associated with the SIS train.

B 3.5.4 Refueling Water Storage Pit (RWSP)

BASES

BACKGROUND

The RWSP supplies borated water to the Safety Injection System (SIS) and the Containment Spray (CS) System during accident conditions.

The RWSP supplies four trains of the SIS and four trains of the CS System through separate, redundant supply headers. A motor operated isolation valve is provided in each header to allow isolation of the RWSP from its associated SIS or CS train if required for maintenance. Use of a single RWSP to supply all four trains of the SI system and CS System is acceptable since the RWSP is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

Design ensures that during a design basis event, the RWSP is replenished with water which has been released to the containment from the RCS sufficient to maintain adequate net positive suction head to the SI and containment spray/ residual heat removal (CS/RHR) pumps throughout the event.

During normal operation in MODES 1, 2, and 3, the SI and CS/RHR pumps are aligned to take suction from the RWSP.

The SIS and CS system pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

This LCO ensures that:

a. Sufficient borated water volume exists to support continued operation of the SI and CS/RHR pumps and

b. The reactor remains subcritical following a LOCA.

Insufficient water in the RWSP could result in insufficient suction head for the SI and CS/RHR pumps. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWSP provides a source of borated water to the SI and CS System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Refs. 1 and 2). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "Safety Injection System (SIS) - Operating," B 3.5.3, " Safety Injection System (SIS) - Shutdown," and B 3.6.6, "Containment Spray ~~and Cooling~~ Systems." These analyses are used to assess changes to the RWSP in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWSP must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWSP, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability.

APPLICABLE SAFETY ANALYSES (continued)

~~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 RWSP 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 limit of 329,150 gallons and the lower boron concentration limit of 4000 ppm are used to compute the post LOCA 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.

The upper limit on boron concentration of 4200 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from direct vessel injection to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWSP lower temperature limit of 32°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 120°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

APPLICABLE SAFETY ANALYSES (continued)

transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWSP water temperature are used to maximize the total energy release to containment.

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

LCO

The RWSP 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 to support SIS and CS/RHR pump operation.

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

APPLICABILITY

In MODES 1, 2, 3, and 4, RWSP OPERABILITY requirements are dictated by the SIS and Containment Spray System OPERABILITY requirements. Since both the SIS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWSP must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1 and A.2

With RWSP boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the SIS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWSP temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time. This Required Action is not applicable in MODE 4.

ACTIONS (continued)

B.1

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

In this Condition, neither the SIS nor the Containment Spray System can 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 RWSP is not required. The short time limit of 1 hour to restore the RWSP to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWSP cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWSP borated water temperature should be verified to be within the limits assumed in the accident analyses band. [The Frequency of 24 hours is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

~~The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWSP. With ambient air temperatures within the band, the RWSP temperature should not exceed the limits.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWSP water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued SI pump and CS/RHR pump operation on recirculation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.3

The boron concentration of the RWSP should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting RWSP pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 6~~Subsection 6.2.2.

2. FSAR ~~Chapter 15~~Subsection 15.6.5.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 pH Adjustment

BASES

BACKGROUND

The Emergency Cooling System (ECCS) includes twenty three NaTB pH adjustment baskets which provide adjustment of the pH of the water in the containment following an accident.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of the ECCS and containment spray system (CSS) components during long-term cooling.

Sodium tetraborate decahydrate (NaTB) contained in baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh that readily permits contact with water. These baskets are located inside three NaTB basket containers at an elevation that is below the lowest spray ring. NaTB in baskets is dissolved in spray water in the containers. The solution containing NaTB is discharged from each container to the RWSP through NaTB solution transfer pipe. Recirculation of water by the safety injection pumps and containment spray / residual heat removal pumps provide mixing to achieve a uniform pH. (Ref. 1)

APPLICABLE SAFETY ANALYSES

In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment is adjusted by the addition of NaTB. Adjusting the water in containment to neutral or alkaline pH (pH ≥ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment. (Ref. 1and 2)

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

The minimum mass of NaTB is 44,100 pounds. This mass is based on providing sufficient NaTB to buffer the post accident containment water to a minimum pH of 7.0. Additionally, the NaTB mass is based on treating the maximum volume of post accident water (879,740 gallons) containing the maximum amount of boron (4200 ppm) as well as other sources of acid.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the NaTB after exposure to the containment environment. A representative sample of 5.71 grams of NaTB from one of the baskets in containment is submerged in ≥ 1liter of water at a boron concentration of 4200 ppm. At the standard temperature of 120 ± 5°F,without agitation, the solution must be left to stand for 12 hours. The liquid is then decaned and mixed, the temperature is adjusted to 77 ± 2°F. At this point, the pH must be ≥ 7.0.

The minimum required amount of NaTB is sufficient to buffer the maximum amount of boron 4200 ppm, the maximum amount of other acids, and the maximum amount of water 879,740 gallons that can exist in the containment following an accident and achieve a minimum pH of 7.0.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 6~~Section 6.3.

2. FSAR ~~Chapter 15~~Subsection 15.6.5.5.

BASES

BACKGROUND (continued)

b. Each airlock is OPERABLE, except as provided in LCO 3.6.2, "Containment Airlocks,"

c. All equipment hatches are closed, and

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and steam line break accidents (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBA involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. In the safety analysis the assumed leakage is 0.15% per day at P_a = 57.5 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

LCO

Containment OPERABILITY is maintained by limiting leakage to. 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR ~~Chapter 15~~Subsection 15.7.4.
3. FSAR ~~Chapter 6~~Subsection 6.2.1.
4. ASME Code, Section XI, Subsection IWL.
5. NEI 94-01.

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 50, Appendix J, (Ref. 1), as L_a , the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_{a,}$ following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.~~3~~4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. FSAR ~~Chapter 6~~Subsection 6.2.1.

BACKGROUND (continued)

High Volume Purge System (36 inch purge valves)

The High Volume Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. The 36 inch purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Low Volume Purge System (8 inch purge valves)

The Low Volume Purge System operates to:

a. Supply outside air into the containment for ventilation and cooling or heating

b. Reduce the concentration of noble gases within containment prior to and during personnel access and

c. Equalize internal and external pressures.

Since the valves used in the Low Volume Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 36 inch high volume purge valves are closed at event initiation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBA analysis assumes that, within ~~15~~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 ~~15~~60 seconds includes signal delay, and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(~~d~~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 36 inch high volume purge valves must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in FSAR Chapter 6 (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.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

BASES

ACTIONS (continued)

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically ~~.~~

~~I~~isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

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

BASES

ACTIONS (continued)

D.1, D.2 and D.3

In the event one or more containment high volume purge valves in one or more penetration flow paths are not within the high volume 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 high volume 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 high volume purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification 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.

For the containment high volume 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 high volume purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 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.

BASES

ACTIONS (continued)

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.

~~D.1 and D.2~~E.1 and E.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 36 inch containment high volume purge valve is required to be verified sealed closed. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment high volume purge valve. Detailed analysis conducted for similar plant design of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment high volume 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, “Containment Leakage Due to Seal Deterioration” (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

~~SR 3.6.3.6~~SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 15~~Subsection 15.6.5.5.

2. FSAR ~~Chapter 6~~Subsection 6.2.4.

3. Standard Review Plan 6.2.4.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

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

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 6~~Subsection 6.2.1.

2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray ~~and Cooling~~ system~~s~~ during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establish the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of two ESF buses, which is the worst case single active failure plus maintenance outage, resulting in two trains each of the Containment Spray/Residual Heat Removal System ~~and Containment Cooling System~~ being rendered inoperable.

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120 °F. This resulted in a maximum containment air temperature of 281.93°F. The containment design temperature is 300°F based on LOCA analysis.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a short time during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the SLB.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray/Residual Heat Removal System (Ref. 1).

The containment pressure transient is sensitive to the temperature of structures in containment which work as heat sinks during the DBAs and therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature. Therefore the containment vessel and required safety related equipment within containment will continue to perform their functions.

APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.
ACTIONS	A.1 When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems. B.1 and B.2 If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.6.5.1 Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. FSAR ~~Chapter 6~~Subsection 6.2.1. 2. CFR 50.49.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," and GDC 40, "Testing of Containment Heat Removal Systems," ~~and GDC 41, "Containment Atmosphere Cleanup,"~~ (Ref. 1).

The Containment Spray System is an Engineered Safety Feature (ESF) system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System limits and maintains post accident conditions to less than the containment design values.

~~Containment Spray System~~

The Containment Spray System consists of four separate trains of equal capacity, ~~each~~ capable of meeting 50% of the design ~~bases~~basis heat removal capacity. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage pit (RWSP) supplies borated water to the Containment Spray System.

The Containment Spray System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. The RWSP solution temperature is an important factor in determining the heat removal capability of the Containment Spray System.

Heat is removed from the ~~containment sump~~RWSP water by the containment spray/residual heat removal ~~coolers~~heat exchangers. Two trains of the Containment Spray System provide adequate spray coverage to meet the system design requirements for containment heat removal.

BASES

BACKGROUND (continued)

The Containment Spray System is actuated either automatically by a High-3 containment pressure signal or manually. An automatic actuation opens the containment spray pump discharge valves and starts the containment spray pumps. A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence. The Containment Spray System maintains an equilibrium temperature between the containment atmosphere and RWSP water. ~~Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.~~

APPLICABLE SAFETY ANALYSES

The Containment Spray System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to Containment integrity are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming one Class 1E bus is out of service and the loss of another Class 1E bus bus, which is the worst case single active failure and results in two trains of Containment Spray System being inoperable.

The analysis and evaluation show that, under the worst case scenario, the highest peak containment pressure is 57.5 psig experienced during a LOCA. The analysis shows that the peak containment temperature is 355°F experienced during an SLB. Both results meet the intent of the design basis. See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, “Containment Temperature” for a detailed discussion. The analyses and evaluations assume a unit specific power level of 100%, two containment spray trains operating, and initial (pre-accident) containment conditions of 120°F and 2 psig. The analyses also assume a response time delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.~~8~~9 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the High-3 containment pressure setpoint to achieving full flow though the containment spray nozzles. The Containment Spray System total response time of 243 seconds includes Class 1E gas turbine generator (GTG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 3).

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

LCO

During a DBA, a minimum of two containment spray trains are required to maintain the containment peak pressure and temperature below the design limits (Ref. 4). To ensure that these requirements are met, three containment spray trains must be OPERABLE. Therefore, in the event of an accident, at least two trains operate, assuming the worst case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWSP upon an ESF actuation signal.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the CS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1 and A.2

If one of the required containment spray trains is inoperable, it must be restored to OPERABLE status within ~~7 days~~72 hours. With a required containment spray train inoperable, the system is capable of providing 100% of the heat removal needs for a DBA . Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a risk informed completion time (RICT). This Required Action is not applicable in MODE 4. The ~~7 day~~72 hours Completion Time was chosen because of the low probability of DBA occurring during this period.

~~B.1~~

~~With one or less containment spray trains OPERABLE, the containment spray system is not capable of providing 100% capacity. Therefore, two trains must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time was chosen as a reasonable time for repairs and low probability of DBA occurring during this period.~~

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

If any of the Required Actions or associated Completion Times for Condition A ~~or B~~ of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated High-3 containment pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

~~The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.~~

SR 3.6.6.5

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. [The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40.
2. 10 CFR 50, Appendix K.
3. FSAR ~~Chapter 15~~Subsection 15.6.5.5.
4. FSAR ~~Chapter 6~~Subsection 6.2.1.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Six MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in ~~the~~ FSAR Chapter 10 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open the valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in ~~the~~ FSAR Chapter 15 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the main steam relief or turbine bypass valves. The FSAR Chapter 15 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if one or more MSSVs on a single steam generator are inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for such case, it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

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

BASES

LCO	The accident analysis requires that six MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that six MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis. The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.
APPLICABILITY	In MODES 1, 2, and 3, six MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.
ACTIONS	The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV. A.1 and A.2 In the case of one or more steam generators with one or more MSSVs inoperable ~~MSSVs per steam generator~~, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Required Action A.1 requires an appropriate reduction in reactor power within 4 hours. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

BASES

ACTIONS (continued)

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for calorimetric power uncertainty.

To determine the maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs, the governing heat transfer relationship is the equation $q = \dot{m}\,\Delta h$, where q is the heat input from the primary side, $\dot{m}$ is the mass flow rate of the steam, and Δh is the increase in enthalpy that occurs in converting the secondary side water to steam. If it is conservatively assumed that the secondary side water is all saturated liquid (i.e., no subcooled feedwater), then the Δh is the heat of vaporization (h_{fg}) at the steam relief pressure. The following equation is used to determine the maximum allowable power level for continued operation with inoperable MSSV(s):

Maximum NSSS Power ≤ $(100/Q)\ (w_s h_{fg} N) / K$

where:

Q=Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

K=Conversion factor, 947.82 (Btu/sec)/MWt

ws=Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest OPERABLE MSSV opening pressure, including tolerance and accumulation, as appropriate, lbm/sec.

hfg=Heat of vaporization at the highest MSSV opening pressure, including tolerance and accumulation as appropriate, Btu/lbm.

N=Number of steam generators in the plant.

To determine the Table 3.7.1-1 Maximum Allowable Power for Required Actions A.1 and A.2 (%RTP), the Maximum NSSS Power calculated using the equation ~~in the previous page~~above is reduced by 9 % RTP to account for Nuclear Instrumentation System trip channel uncertainties.

Required Action A.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 5 inoperable MSSVs per steam generator, 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). According to Reference 5, the following tests are required:

a. Visual examination,

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting),

d. Compliance with owner's seat tightness criteria, and

e. Verification of the balancing device integrity on balanced valves.

The 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. Table 3.7.1-2 allows a ± ~~3~~1% setpoint tolerance for OPERABILITY; ~~however~~and, the valves are reset to remain within ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, 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.

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.3.2.3.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR ~~Chapter 15~~Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM-1-1987.
6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater (EFW) pump turbine steam supply, to prevent MSSV and EFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, turbine bypass system, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by low steam line pressure, steam line pressure negative rate high, or high-high containment pressure. The MSIVs fail closed on loss of control air.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in ~~the~~ FSAR Chapter 10 (Ref. 1). |

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in ~~the~~ FSAR Chapter 6 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in ~~the~~ FSAR Chapter 15 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). |

The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, 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 MSIV

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

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.

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, 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 100 (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, 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.

ACTIONS

A.1 and A.2

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

BASES

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.3.2.3.4.
2. FSAR ~~Chapter 6~~Subsection 6.2.1.
3. FSAR ~~Chapter 15~~Subsection 15.1.5.
4. 10 CFR 100.11.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

BACKGROUND (continued)

All main feedwater valves, MFIVs, MFRVs, MFBRVs, and SGWFCVs close on receipt of any of the following Main Feedwater Isolation signals: high-high steam generator water level, ECCS actuation, or manual actuation.

A description of the MFIVs ,MFRVs, MFBRVs, and SGWFCVs is found in ~~the~~ FSAR Chapter 10 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs ,MFRVs, MFBRVs and SGWFCVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs, MFRVs, MFBRVs, and SGWFCVs, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of high-high steam generator water level signal or a safety injection signal.

Failure of an MFIV, MFRV, MFBRV, or SGWFCV to close following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs, MFRVs, MFBRVs, and SGWFCVs satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO ensures that the MFIVs, MFRVs, MFBRVs, and SGWFCVs will isolate MFW flow to the steam generators, following an FWLB or main steam line break. These valves will also isolate the nonsafety related portions from the safety related portions of the system.

This LCO requires that four MFIVs, four MFRVs, four MFBRVs, and four SGWFCVs be OPERABLE. The MFIVs, MFRVs, MFBRVs, and SGWFCVs are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a ~~safety injection signal~~feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

BASES

reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, MFBRV, and SGWFCV is ≤ 5 seconds. The MFIV, MFRV, MFBRV, and SGWFCV isolation times are assumed in the accident and containment analyses. This Surveillance 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 MFIV, MFRV, MFBRV, and SGWFCV 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.4.7.2.2.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Steam Depressurization Valves (MSDVs)

BASES

BACKGROUND	The MSDVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in ~~the~~ FSAR Chapter 10 (Ref. 1). This is done in conjunction with the Emergency Feedwater System providing cooling water from the emergency feedwater pit (EFP). The MSDVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Turbine Bypass System. One MSDV line for each of the four steam generators is provided. Each MSDV line consists of one MSDV and an associated block valve. The MSDVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The MSDVs are motor operated valves with modulation capability to permit control of the cooldown rate. A description of the MSDVs is found in Reference 1. The MSDVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.
APPLICABLE SAFETY ANALYSES	The design basis of the MSDVs is established by the capability to cool the unit to RHR entry conditions. The design rate of 50°F per hour is applicable for two steam generators, each with one MSDV. This rate is adequate to cool the unit to RHR entry conditions with two steam generators and two MSDVs, utilizing the cooling water supply available in the EFP.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Reference 2, the MSDVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the MSDVs. The number of MSDVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of unit loops and consideration of any single failure assumptions regarding the failure of one MSDV to open on demand.

The MSDVs are equipped with block valves in the event an MSDV fails to close during use.

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

LCO

Four MSDV lines are required to be OPERABLE. One MSDV line is required from each of four steam generators to ensure that at least two MSDV lines are available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second MSDV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open MSDV line.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the turbine bypass system. A closed block valve does not render it or its MSDV line inoperable if operator action time to open the block valve is supported in the accident analysis.

An MSDV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, the MSDVs are required to be OPERABLE.

In MODE4, 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1 and A.2

With one required MSDV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE MSDV lines, a nonsafety grade backup in the turbine bypass system, and MSSVs. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

B.1 and B.2

With two or more MSDV lines inoperable, action must be taken to restore all but one MSDV line to OPERABLE status. Since the block valve can be closed to isolate an MSDV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable MSDV lines, based on the availability of the turbine bypass system and MSSVs, and the low probability of an event occurring during this period that would require the MSDV lines. Required Action B.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

C.1 and C.2

If the MSDV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, within ~~24~~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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the MSDVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the MSDVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an MSDV during a unit cooldown may satisfy this requirement. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.4.2

The function of the block valve is to isolate a failed open MSDV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.3.2.3.3.

2. FSAR ~~Chapter 15~~Subsection 15.6.3.

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater System (EFWS)

BASES

BACKGROUND

The EFWS automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The EFW pumps take suction through separate and independent suction lines from one of the two EFW pits (LCO 3.7.6) and pumps to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam depressurization valves (MSDVs) (LCO 3.7.4).

The EFWS consists of two motor driven EFW pumps and two turbine driven pumps configured into four trains. Each motor driven pump provides 50% of EFW flow capacity, and each turbine driven pump provides 50% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven EFW pump is powered from an independent Class 1E power supply. Each turbine driven EFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven EFW pump.

During normal plant operation (without on-line maintenance (OLM)), all EFW pump discharge cross-connect line isolation valves are closed. Each one of the four EFW pumps is able to supply feedwater separately to each steam generator. During OLM, all the EFW pump discharge cross-connect line isolation valves are opened. Each EFW pump is able to supply feedwater to all steam generators.

The EFWS is capable of supplying feedwater to the steam generators during safe shutdown, transient and accident conditions.

The turbine driven EFW pumps supply EFW using DC powered control valves actuated to the appropriate steam generator by the engineered safety feature actuation system (ESFAS).

Any two of ~~under~~ the four EFW pumps at full flow are sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the EFWS is met.

BACKGROUND (continued)

The EFWS is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs plus 3% margin. Subsequently, the EFWS supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the MSDVs.

The EFWS actuates automatically on low steam generator water level by the ESFAS (LCO 3.3.2). The system also actuates on loss of offsite power, safety injection, and trip of all MFW pumps.

The EFWS is discussed in ~~the~~ FSAR Chapter 10 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EFWS mitigates the consequences of any event with loss of normal feedwater.

The design basis of the EFWS is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% margin.

In addition, the EFWS must supply enough makeup water to replace the secondary steam generator inventory lost as the unit cools to MODE 4 conditions. Sufficient EFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting design basis accidents (DBAs) and transients for the EFWS are as follows:

a. Feedwater line break (FWLB) and

b. Loss of MFW.

In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).

The EFWS design is such that it can perform its function following an FWLB between the MFIVs and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the EFW pump.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EFW flow to the ~~foulty~~faulty steam generator is automatically terminated by the ~~RCPS~~RPS (high SG water level coincident with reactor trip and no low main steam line pressure signal, and at low main steam line pressure) Sufficient flow would be delivered to the intact steam generator by the redundant EFW pump.

The ESFAS automatically actuates the EFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC power operated valves are provided for each EFW line to control the EFW flow to each steam generator.

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

LCO

This LCO provides assurance that the EFWS will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Four independent EFW pumps in four diverse trains are required to be OPERABLE to ensure the availability of RHR capability for all events accompanied by a loss of offsite power and a single failure during non-OLM. This is accomplished by powering two of the pumps by independent emergency buses. The third and fourth EFW pumps are powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

During OLM, three of four independent EFW pumps in three of the four diverse trains are required to be OPERABLE. This LCO may be changed from Required Action A or B when the EFW pump discharge cross-connect line isolation valves are closed if an EFW train becomes inoperable during non-OLM.

The EFWS is configured into four trains. The EFWS is considered OPERABLE when the components and flow paths required to provide redundant EFW flow to the steam generators are OPERABLE. This requires that four EFW pumps be OPERABLE in four diverse paths, each supplying EFW to separate steam generators during non-OLM. During OLM, three FW pumps shall be OPERABLE and shall be capable of supplying EFW to any of supplying EFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

ACTIONS (continued)

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven EFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven EFW pumps, and due to the low probability of an event requiring the use of the turbine driven EFW pump.

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one EFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required EFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore to OPERABLE status within 72 hours or open all EFW pump discharge cross-connect line isolation valves within 72 hours . This Condition includes the loss of two steam supply lines to the turbine driven EFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFWS, time needed for repairs, and the low probability of a DBA occurring during this time period.

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two required EFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~18~~12 hours.

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

In MODE 4 with two required EFW trains inoperable, operation is allowed to continue because only one or two motor driven pump EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

BASES

ACTIONS (continued)

D.1

If three EFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one additional EFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one additional EFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFWS water and steam supply flow paths provides assurance that the proper flow paths will exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. The Note 2 states that one or more EFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the EFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since EFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the EFWS. OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.5

This SR verifies that the EFW is properly aligned by verifying the flow paths from the EFW pits to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFWS during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure EFWS alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the ~~CST~~EFW pits to the steam generators is properly aligned.

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.4.9.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater Pit (EFW Pit)

BASES

BACKGROUND	The two EFW Pits provide safety grade sources of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The EFW Pits provide a passive flow of water, by gravity, to the Emergency Feedwater (EFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the main steam depressurization valves. The EFW pumps operate with a continuous recirculation to the EFW Pits. When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the turbine bypass system. This has the advantage of conserving condensate while minimizing releases to the environment. Because the EFW Pits are principal components in removing residual heat from the RCS, they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The EFW Pits are designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources. A description of the EFW Pits is found in FSAR Chapter 10 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The EFW Pits provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in FSAR Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 2 hours at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate. The limiting event for the EFW Pits volume is the large feedwater line break coincident with a loss of offsite power. The EFW Pits satisfy Criteria 2 and 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

BASES

ACTIONS

A.1, A.2.1, and A.2.2

If the one or both EFW Pits are not OPERABLE, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the EFW pumps are OPERABLE, and that the backup supply has the required volume of water available. Both EFW Pits must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the EFW Pits. Required Action A.2.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

B.1 and B.2

If both EFW Pits cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, within ~~24~~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 3.7.6.1

This SR verifies that each EFW Pit contains the required volume of cooling water. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 10~~Subsection 10.4.9.2.1.
2. FSAR ~~Chapter 6~~Subsection 6.2.1.4.
3. FSAR Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from RHR entry conditions (T_{cold} < 350°F), to MODE 5 (T_{cold} < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and CS/RHR trains operating. Two CCW trains are sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

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

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, two CCW trains are required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, three trains of CCW must be OPERABLE. At least two CCW trains will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCW train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the CS/RHR heat exchanger.

BASES

SURVEILLANCE REQUIREMENTS (continued)

verification that those valves capable of being mispositioned are in the correct position.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 9~~Subsection 9.2.2.

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water System (ESWS)

BASES

BACKGROUND

The ESWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the ESWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ESWS consists of four separate, safety related, cooling water trains. Each train consists of one 50% capacity pump, one component cooling water (CCW) heat exchanger, ~~one Class 1E gas turbine generator (GTG) cooler~~one essential chiller unit, piping, valv~~ing~~es, instrumentation, and two types of strainers. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). 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.

Additional information about the design and operation of the ESWS, along with a list of the components served, is presented in FSAR Chapter 9 (Ref. 1). The principal safety related function of the ESWS is the removal of decay heat from the reactor via the CCW System.

APPLICABLE SAFETY ANALYSES

The design basis of the ESWS is for two ESWS trains, in conjunction with the CCW System to remove core decay heat following a design basis LOCA. This prevents the refueling water storage pit fluid from increasing in temperature following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System. The ESWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ESWS, in conjunction with the CCW System, also cools the unit from containment spray/residual heat removal (CS/RHR), as discussed in FSAR Chapter 5, (Ref. 2) 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 CS/RHR System trains that are operating.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two ESWS trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESWS temperature of 95°F occurring simultaneously with maximum heat loads on the system.

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

LCO

Three of the four ESWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ESWS train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The pump is OPERABLE and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESWS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESWS are determined by the systems it supports.

ACTIONS

A.1 and A.2

If one of the required ESWS trains is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ESWS trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ESWS trains could result in loss of ESWS function. ~~Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Required Action is not applicable in MODE 4.~~ Required Action A.1 is modified by two Notes.

BASES

ACTIONS (continued)

~~The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ESWS train results in an inoperable GTG.~~ The ~~second~~ Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESWS train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Required Action is not applicable in MODE 4. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the ESWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ESWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the ESWS.

Verifying the correct alignment for manual, power operated, and automatic valves in the ESWS flow path provides assurance that the proper flow paths exist for ESWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ESWS valves on an actual or simulated actuation signal. The ESWS is a normally operating system that cannot be fully actuated as part of normal testing. This surveillance is tested to assure the requirements of IST program described in Table 3.9-14. The motor operated valve is provided at the discharge of each pump. The starting logic of the ESWP interlocks the motor operated valve with the pump operation. This interlock prevents the pump from starting if the valve is not closed. The closed discharge valve opens after starting the ESWP. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

This SR verifies proper automatic operation of the ESWS pumps on an actual or simulated actuation signal. The ESWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 9~~Subsection 9.2.1.

2. FSAR ~~Chapter 5~~Subsection 5.4.7.

BACKGROUND (continued)

The air entering the CRE is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the pressurization moded. The actions of the isolation mode are more restrictive, and will override the actions of the pressurization mode.

A single train of MCREFS operating at a flow ≤1200 cfm will pressurize the CRE to about 0.125 inches water gauge relative to external areas adjacent to the CRE boundary. The MCRVS operation in maintaining the CRE habitable is discussed in ~~the~~ FSAR Chapter 9, Subsection 9.4.1 (Ref. 2).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The MCREFS is designed in accordance with Seismic Category I requirements.

Two trains of MCRATCS will provide the required temperature control to maintain the control room between 73°F and 78°F. The MCRVS operation in maintaining the control room temperature is discussed in ~~the~~ FSAR Chapter 9, Section 9.4.1 (Ref. 2).

The CRE habitability is maintained by limiting the inleakage of potentially contaminated air into the CRE. The potential leakage paths for the CRE include the control room enclosure (e.g., walls, penetrations, floor, ceilings, joints, etc.) and other potential paths such as pressurized ductwork from other HVAC systems, pressurized air systems (e.g., instrument air) or isolated HVAC intakes.

The periodic surveillance pressurization tests verify the integrity of the CRE with respect to potentially contaminated adjacent areas. It does not verify filtered inleakage internal to the filtration units and ductwork nor does it verify unfiltered inleakage from internal pressurized sources (e.g., instrument air). These sources of inleakage are addressed separately from TS surveillances.

The MCRVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The MCRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The MCREFS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the design basis accident (DBA), fission product release presented in FSAR Chapter 15, Subsection 15.6.5.5 (Ref. 3).

The MCRATCS maintains the temperature between 73°F and 78°F.

The emergency pressurization mode of the MCRVS is assumed to operate following a DBA to provide protection from a radiological dose to the CRE occupants. The MCRVS also provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 1). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown console. The analysis for Comanche Peak Units 3 and 4 has determined no MCREFS actuation is required based on hazardous chemical releases or smoke and no Surveillance Requirements are required to verify OPERABILITY based on hazardous chemicals or smoke.

The worst case single active failure of a component of the MCRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

LCO

Two independent and redundant MCREFS trains and three of the four independent and redundant MCRATCS are required to be OPERABLE to provide the required redundancy to ensure that the system functions assuming the worst case single active failure occurs coincident with the loss of offsite power. Total system failure, such as from a loss of the required ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the CRE occupants in the event of a large radioactive release and in the equipment operating temperature exceeding limits in the event of an accident.

BASES

APPLICABILITY

In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies, the MCRVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA and to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

During movement of irradiated fuel assemblies, the MCRVS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

When one MCREFS train is inoperable for reasons other than an inoperable CRE boundary (only one MCREFS train OPERABLE), action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE MCREFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a single failure in the OPERABLE MCREFS train could result in loss of function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

When ~~two~~one of the required MCRATCS trains are inoperable (only two MCRATCS trains OPERABLE), action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE MCRATCS trains are adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE MCRATCS trains could result in loss of function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

ACTIONS (continued)

D.1 and D.2

In MODE 1, 2, 3, or 4, if the inoperable MCRVS train or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, 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.

E.1 and E.2

During movement of irradiated fuel assemblies, if the inoperable MCREFS or/and MCRATCS trains cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE MCRVS trains in the emergency mode. This action ensures that the remaining trains are OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action E.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 the accident risk. This does not preclude the movement of fuel to a safe position.

F.1

During movement of irradiated fuel assemblies, with ~~two~~ required MCRVS ~~trains~~ inoperable or with ~~one or more~~ required MCRVS ~~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 require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

G.1

If ~~both~~ required MCRVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition C), the MCRVS 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 dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥10 continuous hours with the heaters energized. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

This SR verifies that the required MCREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each MCRVS train starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. For Comanche Peak there is no MCREFS actuation for hazardous chemical releases or smoke and there are no Surveillance Requirements that verify OPERABILITY for hazardous chemicals or smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition C must be entered.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.4 (continued)

Required Action C.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 5). These compensatory measures may also be used as mitigating actions as required by Required Action C.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 6). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

SR 3.7.10.5

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of a combination of testing and calculations. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 6~~Subsection 6.4.4.

2. FSAR ~~Chapter 9~~Subsection 9.4.1.

3. FSAR ~~Chapter 15~~Subsection 15.6.5.5.

4. Regulatory Guide 1.196

5. NEI 99-03, “Control Room Habitability Assessment,” June 2001.

6. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, “NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternate Source Terms in the Context of Control Room Habitability.” (ADAMS Accession No. ML040300694).

B 3.7 PLANT SYSTEMS

B 3.7.11 Annulus Emergency Exhaust System

BASES

BACKGROUND

The annulus emergency exhaust system is the engineered safety feature (ESF) filter system that is designed for fission product removal and retention by filtering air it exhausts from the

- Penetration areas
- Safeguard component areas

following accidents. The annulus emergency exhaust system is automatically initiated by ECCS actuation signal and initiated manually during non-ECCS actuation events, and will establish and maintain a -1/4 in. W.G. pressure in the penetration and safeguard component area, within approximately 240 seconds. Airborne radioactive material in the penetration and safeguard component areas is directed to the annulus emergency exhaust system, preventing uncontrolled release to the environment. The annulus emergency exhaust system exhaust fans direct flow to the plant vent stack.

The annulus emergency exhaust system consists of two independent and redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter and a fan. Ductwork, dampers and instrumentation also form part of the system. Each train is protected by normally-closed exhaust and outlet dampers. These dampers block auxiliary building HVAC system flow into each train during normal operation, thus preserving and extending the useful service live of annulus air filtration media. The system initiates filtered ventilation upon ECCS actuation. In addition, the signal starting the annulus emergency exhaust system exhaust fans opens the corresponding exhaust damper to the plant vent stack, and the exhaust dampers from the penetration and safeguard component areas.

The auxiliary building HVAC system ventilates the penetration and safeguard component areas during normal plant operations and refueling, with normally-open isolation dampers located in the associated supply and exhaust ducts. These isolation dampers automatically close on an ECCS actuation signal, while motor-operated dampers in parallel exhaust ducts from the penetration areas, and parallel exhaust ducts from the safeguard component areas automatically open. Both annulus emergency exhaust filtration unit fan trains A and B outlet dampers open when their respective fan starts on ECCS actuation. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters.

BASES

BACKGROUND (continued)

The annulus emergency exhaust system is discussed in FSAR ~~Chapters 6~~Subsection 6.5.1 and ~~9~~Subsection 9.4.5 (Refs. 1 and 2 respectively).

APPLICABLE SAFETY ANALYSES	The annulus emergency exhaust system design basis is established by the large break loss of coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as valve packing leakage during a Design Basis Accident (DBA). The system evaluation also assumes a passive failure of the ECCS outside containment, such as an SI pump seal leakage. In such a case, the system restricts the radioactive release to within the 10 CFR 50.34 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 50.34 limits). The analysis of the effects and consequences of a large break LOCA are presented in FSAR Chapter 15, Subsection 15.6.5.5 (Ref. 3). The annulus emergency exhaust system also actuates following a small break LOCA to clean up releases of smaller leaks, such as from valve stem packing. Either a complete loss of function or excessive LEAKAGE may result in less efficient removal of any gaseous or particulate material released to the penetration areas and the ECCS pump rooms following a LOCA. The annulus emergency exhaust system satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	Two independent and redundant trains of the annulus emergency exhaust system are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the penetration and safeguard component areas exceeding 10 CFR 50.34 limits in the event of a Design Basis Accident (DBA). The annulus emergency exhaust system is considered OPERABLE when the individual components necessary to control radioactive releases and maintain the safeguard component areas filtration are OPERABLE in both trains. An annulus emergency exhaust system train is considered OPERABLE when its associated: a. Fan is OPERABLE, b. HEPA filter is not excessively restricting flow, and is capable of performing their filtration function, and c. Ductwork and dampers are OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.2

This SR verifies that the required annulus emergency exhaust system testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter ~~performance~~, and minimum ~~system~~ flow rate. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies that the annulus emergency exhaust system starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.11.4

This SR verifies the integrity of the penetration and safeguard component areas enclosure. The ability of the penetration and safeguard component areas to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of annulus emergency exhaust system. During the accident condition, the annulus emergency exhaust system is designed to maintain a ≤-0.25 inches water gauge relative to atmospheric pressure at a flow rate of 5600 cfm in the associated room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The minimum system flow rate maintains a slight negative pressure in the penetration and safeguard component areas, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about 5600 cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter at the design flow rate. An increase in pressure drop or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 6~~Subsection 6.5.1.
2. FSAR ~~Chapter 9~~Subsection 9.4.5.
3. FSAR ~~Chapter 15~~Subsection 15.6.5.5.
4. 10 CFR 50.34.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Storage Pit Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pit meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pit design is given in ~~the~~ FSAR Chapter 9 (Ref. 1). A description of the Spent Fuel Pit Purification and Cooling System is given in FSAR Chapter 9 (Ref. 1). The assumptions of the fuel handling accident are given in FSAR Chapter 15 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pit meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 3). The resultant 2 hour total effective dose equivalent per person at the exclusion area boundary is a small fraction of the 10 CFR 50.34 (Ref. 4) limits.

According to Reference 3, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pit surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 3 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis conservatively assumes that all fuel rods fail~~, conservatively~~.

The fuel storage pit water level satisfies Criteria 2 and 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The fuel storage pit water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel storage and movement within the fuel storage pit.

BASES

APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the fuel storage pit since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pit water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pit is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.12.1 This SR verifies sufficient fuel storage pit water is available in the event of a fuel handling accident. The water level in the fuel storage pit must be checked at the start of spent fuel movement campaign and periodically. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. During refueling operations, the level in the fuel storage pit is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.
REFERENCES	1. FSAR ~~Chapter 9~~Section 9.1. 2. FSAR ~~Chapter 15~~Subsection 15.7.4. 3. Regulatory Guide 1.183, July 2000. 4. 10 CFR 50.34.

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pit Boron Concentration

BASES

BACKGROUND

The spent fuel pit is a single region spent fuel rack design with 11.1 in center-to-center rack spacing. The spent fuel pit stores 900 spent fuel assemblies in borated water without credit for fuel burnup.

The water in the spent fuel storage pit normally contains soluble boron which results in large subcriticality margins under actual operating conditions. The criticality safety design criteria adopted is 10 CFR 50.68(b)(4) for soluble boron credit. By taking partial credit for soluble boron, Keff does not exceed 0.95 when flooded with borated water and Keff remains below 1.0 (subcritical) when flooded with unborated water.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of the fuel storage pit. Examples of these accident conditions are the straight and inclined dropping of a fuel assembly onto the top of the rack. However, accidents that could increase reactivity can be postulated. This increase in reactivity is unacceptable with unborated water in the storage pit. Thus, for these accident occurrences, the presence of abundant soluble boron in the storage pit prevents criticality. One accident that can be postulated is associated with a fuel assembly which is dropped on or misloaded between fully loaded storage racks. This could have a small positive reactivity effect. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by the postulated accident scenario.

The concentration of dissolved boron in the fuel storage pit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pit boron concentration is required to be ≥ 4000 ppm according to the RWSP and refueling requirements. The specified concentration of dissolved boron in the fuel storage pit preserves the assumptions used in the analyses of the soluble boron credit including the potential critical accident scenarios as described in Reference 1. This concentration of dissolved boron is necessary to control reactivity during fuel assembly storage and movement within the fuel storage pit.

BASES

APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pit until a complete spent fuel storage pit verification has been performed following the last movement of fuel assemblies in the spent fuel storage pit. 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.
ACTIONS	A.1, A.2.1, and A.2.2 The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. When the concentration of boron in the fuel storage pit is less than which is 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 of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pit fuel locations to ensure proper locations of the fuel can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position. 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, the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.13.1 This SR verifies that the concentration of boron in the fuel storage pit is within the required limit. As long as this SR is met, the accidents mentioned are fully addressed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. ~~RSAR Chapter 9~~FSAR Section 9.1.

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

–The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 50.34 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the steam system piping failure, as discussed in the FSAR Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of the steam system piping failure do not exceed a small fraction of the unit EAB limits (Ref. 1) for total effective dose equivalent.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and the main steam relief valves (MSRVs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

B 3.7 PLANT SYSTEMS

B 3.7.15 Main Steam Line Leakage

BASES

BACKGROUND	The purpose of the Main Steam Line Leakage LCO is to limit system operation in the presence of leakage from the main steam line inside containment to amounts that do not compromise safety consistent with the Leak-Before-Break (LBB) analysis discussed in Chapter 3, Section 3.6 (Ref. 1). This LCO specifies the amounts of leakage from the main steam line inside containment. LBB methodology allows elimination of postulated pipe breaks in certain piping systems based on the system characteristics and failure mechanics-based crack growth in conjunction with leak detection capability. As described in Section 3.6 (Ref. 1), the LBB concept is applied to the main steam piping inside containment. This LCO deals with protection of the main steam line inside containment from degradation and helps assure that serious leaks will not develop. The consequences of violating this LCO include the possibility of further degradation of the main steam lines, which may lead to pipe break.
APPLICABLE SAFETY ANALYSES	The safety analyses do not address the main steam line leakage. The safety significance of leakage inside containment varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring leakage into the containment area is necessary. The leakage detection instrumentations required by LCO 3.4.15 perform this function. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public. Although the main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria, this specification has been included in Technical Specifications because the LBB concept is applied to the main steam piping as well as RCL piping.
LCO	Main steam line leakage shall be limited to: 0.5 gallon per minute (gpm) including leakage from the main steam line inside containment is since it is below the leakage rate for LBB analyzed cases of a main steam line crack twice as long as a crack leaking at ten (10) times the detectable leak rate under normal operating load conditions. Violation of this LCO could result in continued degradation of the main steam line.

BASES

APPLICABILITY	In MODES 1, 2, 3, and 4, the potential for main steam line leakage is greatest when the main steam line is pressurized. In MODES 5 and 6, main steam line leakage limits are not required because the main steam line pressure is far lower, resulting in lower stresses and reduced potentials for main steam line leakage.
ACTIONS	A.1 and A.2 If main steam line leakage is not within limit, the unit must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the main steam line leakage and also reduces the factors that tend to degrade the main steam line. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the main steam line are much lower, and further deterioration is much less likely.
SURVEILLANCE REQUIREMENTS	SR 3.7.15.1 Verifying main steam line leakage to be within the LCO limits ensures the integrity of the main steam line inside containment is maintained. Main steam line leakage would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. An early warning of main steam line leakage or unidentified LEAKAGE is provided by the automatic systems that monitor the level of containment sump used to collect unidentified LEAKAGE and air cooler condensate flow rate. These leakage detection systems are specified in LCO 3.4.15, “RCS Leakage Detection Instrumentation.” Also, by performance of an RCS water inventory balance, indication of containment environmental pressures, temperatures and radiation allow the determination of whether the main steam line is a potential source of unidentified LEAKAGE inside containment.
REFERENCES	1. Chapter 3, Section 3.6 “Protection Against Dynamic Effects Associated with Postulated Rupture of Piping.”

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E ac electrical power distribution system ac sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A, B, C, and D Class 1 E Gas Turbine Generators (GTGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the ac electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E ac distribution system is divided into redundant load groups (trains) so that the loss of any one or two groups does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single Class 1E GTG.

Offsite power is supplied to the unit switchyard(s) from the transmission network by two transmission lines. From the switchyard(s), two electrically and physically separated circuits provide ac power, through auxiliary transformers, to the Class 1E 6.9 kV buses. A detailed description of the offsite power network and the circuits to the Class 1E 6.9kV buses is found in FSAR ~~Chapter 8~~ Section 8.2 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E 6.9kV bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E distribution system. Within 3 minutes after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the ac electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least two trains of the onsite or offsite ac sources OPERABLE during accident conditions in the event of

a. An assumed loss of all offsite power or all onsite ac power and

b. A worst case single failure.

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

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E electrical power system and separate and independent Class 1E GTGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after AOO or PA.

Qualified offsite circuits are those that are described in FSAR ~~Chapter 8~~Section 8.2 (Ref. 2) and are part of the licensing basis for the unit.

In addition, one required automatic load sequencer per train must be OPERABLE.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Class 1E 6.9kV buses.

One of two circuits is through the unit auxiliary transformer, and the other circuit is from the reserve auxiliary transformer to each Class 1E 6.9kV bus. Normally Class 1E 6.9kV buses are supplied power from the reserve auxiliary transformer.

Each Class 1E GTG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective Class 1E 6.9kV bus on detection of bus undervoltage. This will be accomplished within 100 seconds. Each Class 1E GTG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the Class 1E 6.9kV buses. These capabilities are required to be met from a variety of initial conditions such as Class 1E GTG in standby with the engine hot and GTG in standby with the engine at ambient conditions. Additional Class 1E GTG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the Class 1E GTG to revert to standby status on an ECCS actuation signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for Class 1E GTG OPERABILITY.

BASES

ACTIONS (continued)

A.2.1 and A.2.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and Class 1E GTGs are adequate to supply electrical power to the onsite Class 1E distribution system.

Required Action A.2.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT).

The 72 hour Completion Time takes into account the capacity and capability of the remaining ac sources, a reasonable time for repairs, and the low probability of PA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable Class 1E GTG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that Class 1E GTGs in two trains are inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven ~~auxiliary~~emergency feedwater pumps. Two train systems, such as turbine driven ~~auxiliary~~emergency feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable Class 1E GTG.

ACTIONS (continued)

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable Class 1E GTG exists and

b. A required feature on the other trains (Train A, B, C or D) is inoperable.

If at any time during the existence of this Condition (Class 1E GTGs in two trains inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering required Class 1E GTGs in two trains inoperable coincident with two or more inoperable required support or supported features, or both, that are associated with the OPERABLE Class 1E GTG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is ~~Acceptable~~acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE Class 1E GTG and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution systems. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining ac sources, a reasonable time for repairs, and the low probability of PA occurring during this period.

BASES

ACTIONS (continued)

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE Class 1E GTG(s). If it can be determined that the cause of the inoperable Class 1E GTG does not exist on the OPERABLE Class 1E GTG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other Class 1E GTG(s), the other Class 1E GTG(s) would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable Class 1E GTG cannot be confirmed not to exist on the remaining Class 1E GTG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that Class 1E GTG.

In the event the inoperable Class 1E GTG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE Class 1E GTGs are not affected by the same problem as the inoperable Class 1E GTGs.

B.4.1 and B.4.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.

Required Action B.4.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT).

In Condition B, the remaining OPERABLE Class 1E GTG and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution systems. The 72 hour Completion Time takes into account the capacity and capability of the remaining ac sources, a reasonable time for repairs, and the low probability of PA occurring during this period.

C.1, C.2.1 and C.2.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions.

ACTIONS (continued)

The Class 1E GTGs are adequate to supply electrical power to the onsite Class 1E Distribution System. The 12 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that safety trains are completely OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal “time zero” for beginning the allowed outage time “clock.” In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite ac sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two ac sources inoperable that involve one or more Class 1E GTGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant ac electrical power system that remains available is not susceptible to a single bus or switching failure and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite ac source.

ACTIONS (continued)

With both of the required offsite circuits inoperable, sufficient onsite ac sources are available to maintain the unit in a safe shutdown condition in the event of AOO or PA. In fact, a simultaneous loss of offsite ac sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an ac electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite ac sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A. The Class 1E GTGs connect to Class 1E buses when other all ac power sources are unavailable, until one required offsite circuit is restored to OPERABLE status. COMPLETION TIME of subsequent CONDITION is limited by maximum COMPLETION TIME in accordance with administrative control.

Required Action C.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT)

D.1, D.2 and D.3

Pursuant to LCO 3.0.6, the distribution system ACTIONS would not be entered even if all ac sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no ac source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one or two Class 1E GTG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR ~~Chapter 8~~Section 8.2.
3. Regulatory Guide 1.9, Rev. 4, March 2007.
4. FSAR Chapter 6.
5. FSAR Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.137, Rev.1, October 1979.
10. ASME Code for Operation and Maintenance of Nuclear Power Plants.
11. IEEE Standard 308-2001.

APPLICABLE SAFETY ANALYSES (continued)

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite Class 1E Gas Turbine Generator (GTGs) power.

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

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. OPERABLE Class 1E GTGs, associated with distribution systems trains required to be OPERABLE by LCO 3.8.10, ensure a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit(s) and Class 1E GTGs ensures the availability of sufficient ac sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Class 1E bus(es). Qualified offsite circuits are those that are described in FSAR ~~Chapter 8~~Section 8.2 (Ref. 1) and are part of the licensing basis for the unit.

One of two circuits is through the unit auxiliary transformer, and the other circuit is from the reserve auxiliary transformer to each Class 1E bus. Normally Class 1E 6.9kV buses are supplied power from the reserve auxiliary transformer.

BASES

ACTIONS (continued)

Pursuant to LCO 3.0.6, the distribution system's ACTIONS would not be entered even if all ac sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no ac power to any required Class 1E bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the ac sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.7 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.11 and SR 3.8.1.18 are not required to be met because the ESF actuation signal is not required to be OPERABLE. SR 3.8.1.16 is not required to be met because the required OPERABLE Class 1E GTG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.19 is excepted because starting independence is not required with the Class 1E GTG(s) that is not required to be operable.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE Class 1E GTG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required Class 1E 6.9 kV bus or disconnecting a required offsite circuit during performance of SRs. With limited ac sources available, a single event could compromise both the required circuit and the Class 1E GTGs. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the Class 1E GTGs and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

1. FSAR ~~Chapter 8~~Section 8.2.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Class 1E Gas Turbine Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each Class 1E Gas Turbine Generator (GTG) is provided with a storage tank having a fuel oil capacity sufficient to operate that gas turbine for a period of 7 days while the Class 1E GTG is supplying maximum post loss of coolant accident load demand discussed in FSAR ~~Chapter 9~~Subsection 9.5.4 (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any four Class 1E GTGs is available. This onsite fuel oil capacity is sufficient to operate the Class 1E GTGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. All outside tanks, pumps, and piping are located underground.

For proper operation of the standby Class 1E GTGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The Class 1E GTG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated Class 1E GTG under all loading conditions. The system is required to circulate the lube oil to the gas turbine engine working surfaces and to remove excess heat generated by friction during operation. The engine oil sump in each Class 1E GTG gear boxes contains an inventory capable of supporting a minimum of 7 days of operation. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

Each Class 1E GTG has an air start system with adequate capacity for three successive start attempts on the Class 1E GTG without recharging the air start receiver(s).

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in FSAR Chapter 6 (Ref. 4), and in FSAR Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E GTGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for FSAR Section 3.2, Power Distribution Limits; FSAR Section 3.4, Reactor Coolant System (RCS); and FSAR Section 3.6, Containment Systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Since gas turbine fuel oil, lube oil, and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Stored gas turbine fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of Class 1E GTGs required to shut down the reactor and to maintain it in a safe condition for AOO or PA with loss of offsite power. Class 1E GTG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system is required to have a minimum capacity for three successive Class 1E GTG start attempts without recharging the air start receivers.

APPLICABILITY

The ac sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after AOO or PA. Since stored gas turbine fuel oil, lube oil, and the starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored Class 1E GTG fuel oil, lube oil, and starting air are required to be within limits when the associated Class 1E GTG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each Class 1E GTG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable Class 1E GTG subsystem. Complying with the Required Actions for one inoperable Class 1E GTG subsystem may allow for continued operation, and subsequent inoperable Class 1E GTG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

A.1

In this Condition, the 7 day fuel oil supply for a Class 1E GTG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the Class 1E GTG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. Fuel oil storage tank inventory is controlled to restore following any operation of a gas turbine, including maintenance and testing runs by administrative control.

B.1

With lube oil inventory < 81 gallons, sufficient lubricating oil to support 7 days of continuous Class 1E GTG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the Class 1E GTG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

ACTIONS (continued)

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the gas turbine engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated Class 1E GTG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the Class 1E GTG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a Class 1E GTG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the Class 1E GTG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < ~~270~~398 psig, sufficient capacity for three successive Class 1E GTG start attempts does not exist. However, as long as the receiver pressure is > ~~185~~228 psig, there is adequate capacity for at least one start attempt, and the Class 1E GTG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the Class 1E GTG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most Class 1E GTG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

BASES

REFERENCES

1. FSAR ~~Chapter 9~~Subsection 9.5.4.

2. Regulatory Guide 1.137, Rev.1, October 1979.

3. ANSI N195-1976, Appendix B.

4. FSAR Chapter 6.

5. FSAR Chapter 15.

6. ASTM Standards: D4057-06; D975-07b; D1298-99 (Reapproved 2005); D4176-04^{E1}; D2709-96 (Reapproved 2006); D1552-03; D2622-07; D4294-03; D5452-06.

7. ASTM Standards, D975-07b, Table 1.

BACKGROUND (continued)

Each battery has adequate storage capacity to meet the duty cycle(s) discussed in FSAR ~~Chapter 8~~Subsection 8.3.2 (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for Train A, B, C, and D dc electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 108 V.

The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 125 V for a 60 cell battery (i.e., cell voltage of 2.065 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with its open circuit voltage ≥ 2.065 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. Optimal long term performance however, is obtained by maintaining a float voltage 2.17 to 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.07 Vpc corresponds to a total float voltage output of 124.2 V for a 60 cell battery.

Each Train A, B, C, and D dc electrical power subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient excess capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in FSAR ~~Chapter 8~~Subsection 8.3.2 (Ref. 4).

The battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

When desired, the charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then

BASES

BACKGROUND (continued)

reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in FSAR Chapter 6 (Ref. 5) and FSAR Chapter 15 (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The dc electrical power systems provide normal and emergency dc electrical power for the Class 1E GTGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the dc sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the dc sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite ac power or all onsite ac power and

b. A worst-case single failure.

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

LCO

The dc electrical power subsystems, each subsystem consisting of one battery, battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated bus. This LCO requires three trains to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after AOO or PA. Loss of any train dc electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE dc electrical power subsystem requires the battery and its respective charger to be operating and connected to the associated dc bus.

ACTIONS

A.1, A.2, and A.3

Condition A represents two trains with battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability. The minimum established float voltage will be selected from manufacturer's recommendation float voltage range and controlled by administrative control.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the dc system is designed for.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated dc subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (2.17 Vpc or 130.2 V at the battery terminals). This voltage is controlled by administrative control. This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 800 amps at the minimum established float voltage for 8 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of ac power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional

BASES

SURVEILLANCE REQUIREMENTS (continued)

Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, Rev.0, March 1971.
3. IEEE-308-2001.
4. FSAR ~~Chapter 8~~Subsection 8.3.2.
5. FSAR Chapter 6.
6. FSAR Chapter 15.
7. Regulatory Guide 1.93, Rev.0, December 1974.
8. IEEE-450-2002.
9. Regulatory Guide 1.32, Rev.3, March 2004.
10. Regulatory Guide 1.129, Rev.2, February 2007.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case PA which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

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

LCO

The dc electrical power is made up of subsystems. Each subsystem consists of one battery, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train. A subsystem is required to be OPERABLE to support the associated required trains of the distribution systems specified by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient dc electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The dc electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core,

b. Required features needed to mitigate a fuel handling accident are available,

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Battery parameters satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required dc power to shut down the reactor and maintain it in a safe condition after Anticipated Operational Occurrence (AOO) or PA. Battery parameter limits are conservatively established, allowing continued dc electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the licensee controlled program is conducted as specified in Specification 5.5.17.

APPLICABILITY

The battery parameters are required solely for the support of the associated dc electrical power subsystems. Therefore, battery parameter limits are only required when the dc power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A.1, A.2, and A.3

With one or more cells in one battery in one train < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one battery < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is not assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

ACTIONS (continued)

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one battery in one train with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established. The minimum established design limits are determined based on manufacturer’s recommendation and controlled by administrative control.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.17, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.17.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE-450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the battery may have to be declared inoperable and the affected cells replaced.

D.1

With one or more batteries in one train with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met. The minimum established design limits are determined based on manufacturer’s recommendation and controlled by administrative control.

BASES

ACTIONS (continued)

E.1

With one or more batteries in redundant trains with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one train within 2 hours. The battery parameters limits are determined based on manufacturer's recommendation and controlled by administrative control.

F.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries in one train with one or more battery cells float voltage less than 2.07 V and float current greater than 5 amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTIONS A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 5 amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

SR 3.8.6.2 and SR 3.8.6.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to ~~130.5~~135 V at the battery terminals, or 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.17. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The minimum established design limits are determined based on manufacturer's recommendation and controlled by administrative control.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 40°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The minimum established design limits are determined based on manufacturer's recommendation and controlled by administrative control.

SR 3.8.6.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit. Manufacturer's rating of the battery capacity for an acceptance criterion is determined based on manufacturer's recommendation and controlled by administrative control.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes.

These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND	The inverters are the preferred source of power for the ac vital buses because of the stability and reliability they achieve. The function of the inverter is to provide ac electrical power to the vital buses. The inverters can be powered from an internal ac source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in FSAR ~~Chapter 8~~Subsection 8.3.1 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in FSAR Chapter 6 (Ref. 2) and FSAR Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for FSAR Section 3.2, Power Distribution Limits; FSAR Section 3.4, Reactor Coolant System (RCS); and FSAR Section 3.6, Containment Systems. The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required ac vital buses OPERABLE during accident conditions in the event of: a. An assumed loss of all offsite ac electrical power or all onsite ac electrical power and b. A worst case single failure. Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	The inverters ensure the availability of ac electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after AOO or PA. Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The inverters ensure an

BASES

ACTIONS (continued)

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital bus is re-energized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the ac vital bus is powered from its transformer, it is relying upon interruptible ac electrical power sources (offsite and onsite). The uninterruptible inverter source to the ac vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and ac vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the ac vital buses. The proper voltage and frequency of inverter and breaker alignment are controlled by administrative control. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES

1. FSAR ~~Chapter 8~~Subsection 8.3.1.

2. FSAR Chapter 6.

3. FSAR Chapter 15.

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case PA which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of ac electrical power to the ac vital buses even if the normal power from the 480Vac safety buses are de-energized. OPERABILITY of the inverters requires that the ac vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident are available,

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

ACTIONS (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and ac vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the ac vital buses. The proper voltage and frequency of inverter and breaker alignment are controlled by administrative control. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR Chapter 6.
2. FSAR Chapter 15.

BASES

BACKGROUND (continued)

The list of all required dc and vital ac distribution buses are presented in Table B 3.8.9-1. Specific details on inverters and their operating characteristics are found in FSAR Chapter 8 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in FSAR Chapter 6 (Ref. 1), and in FSAR Chapter 15 (Ref. 2), assume Engineered Safety Features (ESF) systems are OPERABLE. The ac, dc, and ac vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for FSAR Section 3.2, Power Distribution Limits; FSAR Section 3.4, Reactor Coolant System (RCS); and FSAR Section 3.6, Containment Systems.

The OPERABILITY of the ac, dc, and ac vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite ac electrical power and

b. A worst case single failure.

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

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of ac, dc, and ac vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after AOO or PA. The ac, dc, and ac vital bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the required ac, dc, and ac vital bus electrical power distribution subsystems OPERABLE per Table 3.8.9-1, ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the ac, dc, and ac vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The ac, dc, and ac vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the ac, dc, and ac vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum ac, dc, and ac vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The ac and dc electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). The necessary portions in each mode are controlled by administrative control.

APPLICABILITY

The ac and dc electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident are available,

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The ac, dc, and ac vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9, “Distribution Systems - Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pit, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR ~~Chapter 15~~Subsection 15.4.6.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves

BASES

BACKGROUND

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.2.1 These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. FSAR ~~Chapter 15~~Subsection 15.4.6. 2. NUREG-0800, Section 15.4.6.

ACTIONS (continued)

C.1

With no audible alarm and count rate OPERABLE, prompt and definite indication of a boron dilution event, consistent with the assumptions of the safety analysis, is lost. In this situation, the boron dilution event may not be detected quickly enough to assure sufficient time is available for operators to manually isolate the unborated water source and stop the dilution prior to the loss of SHUTDOWN MARGIN. Therefore, action must be taken to prevent an inadvertent boron dilution event from occurring. This is accomplished by isolating all the unborated water flow paths to the Reactor Coolant System. Isolating these flow paths ensures that an inadvertent dilution of the reactor coolant boron concentration is prevented. The Completion Time of "Immediately" assures a prompt response by operations and requires an operator to initiate actions to isolate an affected flow path immediately. Once actions are initiated, they must be continued until all the necessary flow paths are isolated or the circuit is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL 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 CHANNEL CALIBRATION also includes verification of the audible alarm and count rate function. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES	1.	10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
	2.	FSAR ~~Chapter 15~~Subsection 15.4.6.

BASES

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 24 hours prior to irradiated fuel movement without containment closure capability, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.34. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 50.34 to be 25% or less of the 10 CFR 50.34 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 50.34 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 50.34 limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge penetrations. For the OPERABLE containment purge penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge isolation valve closure times specified in the safety analysis can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

Both containment personnel air lock doors may be open during movement of irradiated fuel or CORE ALTERATION, provided an air lock door is capable of being closed and the water level in the refueling pool is maintained as required. Administrative controls ensure that: 1) appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS, 2) specified individuals are designated and readily available to close the air lock following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open air lock can be quickly removed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.3

This Surveillance demonstrates that each containment purge isolation valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handlingaccident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. FSAR ~~Chapter 15~~Subsection 15.7.4.

3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND	The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the Containment Spray (CS)/RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the CS/RHR heat exchanger(s) and the bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.
APPLICABLE SAFETY ANALYSES	While there is no explicit analysis assumptions for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of refueling cavity water level. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE and at least one train of RHR System is operating in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the CS/RHR pumps to be removed from operation for short durations, under the condition that the boron concentration is not reduced. This conditional stopping of the CS/RHR pumps does not result in a challenge to the fission product barrier. RHR and Coolant Circulation – High Water Level satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	Only two RHR loops are required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only two RHR loops are required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least two RHR loops must be OPERABLE and in operation to provide: a. Removal of decay heat,

BASES

ACTIONS (continued)

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 Purge and Exhaust Isolation System.

With 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 3.9.5.1

This Surveillance demonstrates that the RHR loops are in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 5~~Subsection 5.4.7.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the Containment Spray (CS)/RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the CS/RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

While there is no explicit analysis assumptions for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of refueling cavity water level. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Three trains of the RHR System are required to be OPERABLE, and two trains in operation, in order to prevent this challenge.

RHR and Coolant Circulation – Low Water Level satisfies Criterion 4 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, three RHR loops must be OPERABLE. Additionally, two loops of RHR must be in operation in order to provide:

a. Removal of decay heat,

b. Mixing of borated coolant to minimize the possibility of criticality, and

c. Indication of reactor coolant temperature.

BASES

LCO (continued)

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for stopping all RHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loops are OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

An OPERABLE RHR loop consists of an CS/RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

All RHR pumps may be aligned to the Refueling Water Storage Pit to support filling or draining the refueling cavity or for performance of required testing.

APPLICABILITY

Three RHR loops are required to be OPERABLE, and two RHR loops must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. 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 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.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore two RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of three OPERABLE RHR loops and at least ~~one~~two operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

If no RHR 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 Purge and Exhaust Isolation System.

With 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that two RHR loops are in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loops with the water level in the vicinity of the reactor vessel nozzles, the CS/RHR pump suction requirements must be met. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 5~~Subsection 5.4.7.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pit. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within values reported in FSAR Chapter 15.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Appendix B2 of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 3). Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as calculated in Reference 2.

BASES

APPLICABILITY	LCO 3.9.7 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 being moved 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 pit are covered by LCO 3.7.12, "Fuel Storage Pit Water Level."
ACTIONS	A.1 With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving or 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 3.9.7.1 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. Regulatory Guide 1.183, July 2000. 2. FSAR ~~Chapter 15~~Subsection 15.7.4. 3. 10CFR 50.34

B 3.9 REFUELING OPERATIONS

B 3.9.8 Decay Time

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment or in the fuel handling area requires allowing at least 24 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that suffiecient radioactive decay has occurred in the event of a fuel handling 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 handling area, as postulated by Regulatory Guide 1.183 (Ref. 1) The fuel handling accident analysis inside containment or in the fuel handling area is described in Reference 2. This analysis assumes a minimum radioactive decay time of 24 hours. Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum radioactive decay time of 24 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area 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 handling area. 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 inside containment or in the fuel handling area are also covered by LCO 3.7.12, “Fuel Storage Pit Water Level” and LCO 3.9.7, “Refueling Caivty Water Level.”

BASES

ACTIONS	A.1 With a decay time of less than 24 hours, all operations involving movement of irradiated fuel assemblies within containment or in the fuel handling area 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	SR 3.9.8.1 Verification that the reactor has been subcritical for at least 24 hours prior to movement of irradiated fuel in the reactor pressure veseel to the refueling cavity in containment or to the fuel handling area 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, "Alternative Radiological Source Terms Evaluating Design Basis Accidents at Nuclear Power Reactors." 2. FSAR Subsection 15.7.4.