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January 5, 2001

U. S. Nuclear Regulatory Commission
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Gentlemen:

ULNRC-04359

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
TECHNICAL SPECIFICATION BASES REVISION 1

Furnished herewith are the signed original and 10 copies of Revision 1 to the Callaway Plant Technical Specification Bases in accordance with 10 CFR 50.4(b)(6).

Pursuant to 10 CFR 50.71(e), the Technical Specification Bases has been revised to include all of the change made since our revision 0 issue, May 29, 1999, to December 15, 2000.

If there are any questions, please contact us.

Very truly yours,

Alan C. Passwater
Manager, Corporate Nuclear Services

BFH/jdg

Enclosure: Directions for Replacement Pages
Attachment: Revision 1 to Callaway Plant Technical Specification Bases

A001 1/11

STATE OF MISSOURI)
)   S S
CITY OF ST. LOUIS)

Alan C. Passwater, of lawful age, being first duly sworn upon oath says that he is Manager, Corporate Nuclear Services for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C Passwater

Alan C. Passwater
Manager, Corporate Nuclear Services

SUBSCRIBED and sworn to before me this 5th day

of January, 2001.

Melissa L. Orr

MELISSA L. ORR
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My Commission Expires: June 23, 2003

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December 15, 2000

DIRECTIONS FOR INSERTING REVISION 1 TO TS BASES

Remove and insert pages as listed below. Dashes (---) in the remove or insert column of the directions indicate no actions required.

REMOVE		INSERT
	LIST OF EFFECTIVE PAGES	
1, Revision 0i		1, Revision 1
2, Revision 0i		2, Revision 1
3, Revision 0i		3, Revision 1
4, Revision 0i		4, Revision 1
5, Revision 0i		5, Revision 1
6, Revision 0i		6, Revision 1
7, Revision 0i		7, Revision 1
8, Revision 0i		8, Revision 1
9, Revision 0i		9, Revision 1
10, Revision 0i		10, Revision 1
11, Revision 0i		---
12, Revision 0i		---
	BASES PAGES	
	Chapter 3.1 Bases	
B 3.1.1-2, Revision 0a		B 3.1.1-2, Revision 1
B 3.1.2-2, Revision 0a		B 3.1.2-2, Revision 1
B 3.1.3-2, Revision 0a		B 3.1.3-2, Revision 1
B 3.1.5-1, Revision 0a		B 3.1.5-1, Revision 1
B 3.1.6-1, Revision 0a		B 3.1.6-1, Revision 1
B 3.1.7-4, Revision 0a		B 3.1.7-4, Revision 1
B 3.1.7-5, Revision 0a		B 3.1.7-5, Revision 1
B 3.1.7-6, Revision 0a		B 3.1.7-6, Revision 1
B 3.1.7-7, Revision 0a		B 3.1.7-7, Revision 1
B 3.1.8-6, Revision 0a		B 3.1.8-6, Revision 1
	Chapter 3.2 Bases	
B 3.2.3-2, Revision 0a		B 3.2.3-2, Revision 1

December 15, 2000

DIRECTIONS FOR INSERTING REVISION 1 TO TS BASES

Chapter 3.3 Bases

REMOVE	INSERT
B 3.3.1-8, Revision 0a	B 3.3.1-8, Revision 1
B 3.3.1-10, Revision 0a	B 3.3.1-10, Revision 1
B 3.3.1-11, Revision 0a	B 3.3.1-11, Revision 1
B 3.3.1-12, Revision 0a	B 3.3.1-12, Revision 1
B 3.3.1-15, Revision 0a	B 3.3.1-15, Revision 1
B 3.3.1-30, Revision 0a	B 3.3.1-30, Revision 1
B 3.3.1-33, Revision 0a	B 3.3.1-33, Revision 1
B 3.3.1-46, Revision 0a	B 3.3.1-46, Revision 1
B 3.3.1-52, Revision 0a	B 3.3.1-52, Revision 1
B 3.3.1-53, Revision 0a	B 3.3.1-53, Revision 1
B 3.3.1-54, Revision 0b	B 3.3.1-54, Revision 1
B 3.3.1-55, Revision 0b	B 3.3.1-55, Revision 1
B 3.3.1-56, Revision 0b	B 3.3.1-56, Revision 1
B 3.3.1-57, Revision 0b	B 3.3.1-57, Revision 1
B 3.3.1-58, Revision 0b	B 3.3.1-58, Revision 1
B 3.3.1-59, Revision 0a	B 3.3.1-59, Revision 1
B.3.3.1-60, Revision 0a	B.3.3.1-60, Revision 1
B.3.3.1-61, Revision 0a	B.3.3.1-61, Revision 1
B 3.3.2-51, Revision 0a	B 3.3.2-51, Revision 1
B 3.3.2-52, Revision 0a	B 3.3.2-52, Revision 1
B 3.3.2-53, Revision 0a	B 3.3.2-53, Revision 1
B 3.3.2-54, Revision 0a	B 3.3.2-54, Revision 1
B 3.3.3-16, Revision 0a	B 3.3.3-16, Revision 1
B 3.3.3-17, Revision 0a	B 3.3.3-17, Revision 1
B 3.3.4-3, Revision 0a	B 3.3.4-3, Revision 1
B 3.3.4-6, Revision 0a	B 3.3.4-6, Revision 1
B 3.3.6-1, Revision 0a	B 3.3.6-1, Revision 1
B 3.3.6-2, Revision 0a	B 3.3.6-2, Revision 1
B 3.3.6-3, Revision 0a	B 3.3.6-3, Revision 1
B 3.3.6-4, Revision 0a	B 3.3.6-4, Revision 1
B 3.3.6-5, Revision 0a	B 3.3.6-5, Revision 1
B 3.3.6-6, Revision 0a	B 3.3.6-6, Revision 1
B 3.3.7-1, Revision 0a	B 3.3.7-1, Revision 1
B 3.3.7-2, Revision 0a	B 3.3.7-2, Revision 1

December 15, 2000

DIRECTIONS FOR INSERTING REVISION 1 TO TS BASES

REMOVE		INSERT
	Chapter 3.3 Bases (Cont.)	
B 3.3.7-3, Revision 0a		B 3.3.7-3, Revision 1
B 3.3.7-4, Revision 0a		B 3.3.7-4, Revision 1
B 3.3.7-5, Revision 0a		B 3.3.7-5, Revision 1
B 3.3.7-6, Revision 0a		B 3.3.7-6, Revision 1
B 3.3.7-7, Revision 0a		B 3.3.7-7, Revision 1
B 3.3.8-1, Revision 0a		B 3.3.8-1, Revision 1
B 3.3.8-2, Revision 0a		B 3.3.8-2, Revision 1
B 3.3.8-3, Revision 0a		B 3.3.8-3, Revision 1
B 3.3.8-4, Revision 0a		B 3.3.8-4, Revision 1
B 3.3.8-6, Revision 0a		B 3.3.8-6, Revision 1
B 3.3.9-2, Revision 0a		B 3.3.9-2, Revision 1
B 3.3.9-6, Revision 0a		B 3.3.9-6, Revision 1
B 3.3.9-7, Revision 0a		B 3.3.9-7, Revision 1
	Chapter 3.4 Bases	
B 3.4.12-10, Revision 0a		B 3.4.12-10, Revision 1
B 3.4.14-2, Revision 0a		B 3.4.14-2, Revision 1
B 3.4.16-1, Revision 0a		B 3.4.16-1, Revision 1
B 3.4.16-2, Revision 0a		B 3.4.16-2, Revision 1
	Chapter 3.5 Bases	
B 3.5.2-10, Revision 0a		B 3.5.2-10, Revision 1
B 3.5.2-11, Revision 0a		B 3.5.2-11, Revision 1
B 3.5.2-12, Revision 0a		B 3.5.2-12, Revision 1
	Chapter 3.6 Bases	
B 3.6.1-4, Revision 0a		B 3.6.1-4, Revision 1
B 3.6.2-1, Revision 0a		B 3.6.2-1, Revision 1
B 3.6.2-4, Revision 0a		B 3.6.2-4, Revision 1
B 3.6.6-9, Revision 0a		B 3.6.6-9, Revision 1
B 3.6.6-10, Revision 0a		B 3.6.6-10, Revision 1
B 3.6.6-11, Revision 0a		B 3.6.6-11, Revision 1
B 3.6.7-1, Revision 0a		B 3.6.7-1, Revision 1

December 15, 2000

DIRECTIONS FOR INSERTING REVISION 1 TO TS BASES

REMOVE	INSERT
Chapter 3.7 Bases	
B 3.7.1-2, Revision 0a	B 3.7.1-2, Revision 1
B 3.7.1-3, Revision 0a	B 3.7.1-3, Revision 1
B 3.7.1-4, Revision 0a	B 3.7.1-4, Revision 1
B 3.7.1-5, Revision 0a	B 3.7.1-5, Revision 1
B 3.7.3-2, Revision 0a	B 3.7.3-2, Revision 1
B 3.7.3-3, Revision 0a	B 3.7.3-3, Revision 1
B 3.7.3-4, Revision 0a	B 3.7.3-4, Revision 1
B 3.7.3-5, Revision 0a	B 3.7.3-5, Revision 1
B 3.7.4-1, Revision 0a	B 3.7.4-1, Revision 1
B 3.7.4-2, Revision 0a	B 3.7.4-2, Revision 1
B 3.7.4-4, Revision 0a	B 3.7.4-4, Revision 1
B 3.7.5-1, Revision 0a	B 3.7.5-1, Revision 1
B 3.7.5-4, Revision 0a	B 3.7.5-4, Revision 1
B 3.7.5-5, Revision 0a	B 3.7.5-5, Revision 1
B 3.7.5-6, Revision 0a	B 3.7.5-6, Revision 1
B 3.7.5-7, Revision 0a	B 3.7.5-7, Revision 1
B 3.7.5-8, Revision 0a	B 3.7.5-8, Revision 1
B 3.7.5-9, Revision 0a	B 3.7.5-9, Revision 1
B 3.7.5-10, Revision 0a	B 3.7.5-10, Revision 1
B 3.7.7-4, Revision 0a	B 3.7.7-4, Revision 1
B 3.7.8-4, Revision 0a	B 3.7.8-4, Revision 1
B 3.7.10-2, Revision 0a	B 3.7.10-2, Revision 1
B 3.7.10-3, Revision 0a	B 3.7.10-3, Revision 1
B 3.7.10-6, Revision 0a	B 3.7.10-6, Revision 1
B 3.7.11-1, Revision 0a	B 3.7.11-1, Revision 1
B 3.7.11-3, Revision 0a	B 3.7.11-3, Revision 1
B 3.7.13-3, Revision 0a	B 3.7.13-3, Revision 1
B 3.7.15-1, Revision 0a	B 3.7.15-1, Revision 1
B 3.7.15-2, Revision 0a	B 3.7.15-2, Revision 1
B 3.7.15-3, Revision 0a	B 3.7.15-3, Revision 1
B 3.7.16-1, Revision 0a	B 3.7.16-1, Revision 1
B 3.7.16-2, Revision 0a	B 3.7.16-2, Revision 1
B 3.7.17-1, Revision 0a	B 3.7.17-1, Revision 1
B 3.7.18-1, Revision 0a	B 3.7.18-1, Revision 1

December 15, 2000

DIRECTIONS FOR INSERTING REVISION 1 TO TS BASES

REMOVE	INSERT
Chapter 3.8 Bases	
B 3.8.3-3, Revision 0a	B 3.8.3-3, Revision 1
B 3.8.3-4, Revision 0a	B 3.8.3-4, Revision 1
B 3.8.3-5, Revision 0a	B 3.8.3-5, Revision 1
B 3.8.3-6, Revision 0a	B 3.8.3-6, Revision 1
B 3.8.3-7, Revision 0a	B 3.8.3-7, Revision 1
B 3.8.3-8, Revision 0a	B 3.8.3-8, Revision 1
B 3.8.3-9, Revision 0a	B 3.8.3-9, Revision 1
B 3.8.4-8, Revision 0a	B 3.8.4-8, Revision 1
B 3.8.5-4, Revision 0a	B 3.8.5-4, Revision 1
B 3.8.8-3, Revision 0a	B 3.8.8-3, Revision 1
Chapter 3.9 Bases	
B 3.9.1-2, Revision 0a	B 3.9.1-2, Revision 1
B 3.9.1-3, Revision 0a	B 3.9.1-3, Revision 1
B 3.9.1-4, Revision 0a	B 3.9.1-4, Revision 1
B 3.9.3-1, Revision 0b	B 3.9.3-1, Revision 1
B 3.9.3-3, Revision 0a	B 3.9.3-3, Revision 1
B 3.9.5-2, Revision 0a	B 3.9.5-2, Revision 1
B 3.9.5-3, Revision 0a	B 3.9.5-3, Revision 1

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Table of Contents		Chapter 3.1 Bases	
1	0	B 3.1.1-1	0
2	0	B 3.1.1-2	1
3	0	B 3.1.1-3	0
4	0	B 3.1.1-4	0
		B 3.1.1-5	0
		B 3.1.1-6	0
Chapter 2.0 Bases		B 3.1.2-1	0
		B 3.1.2-2	1
B 2.1.1-1	0	B 3.1.2-3	0
B 2.1.1-2	0	B 3.1.2-4	0
B 2.1.1-3	0	B 3.1.2-5	0
B 2.1.1-4	0	B 3.1.3-1	0
B 2.1.2-1	0	B 3.1.3-2	1
B 2.1.2-2	0	B 3.1.3-3	0
B 2.1.2-3	0	B 3.1.3-4	0
		B 3.1.3-5	0
		B 3.1.3-6	0
Chapter 3.0 Bases		B 3.1.3-7	0
		B 3.1.4-1	0
B 3.0-1	0	B 3.1.4-2	0
B 3.0-2	0	B 3.1.4-3	0
B 3.0-3	0	B 3.1.4-4	0
B 3.0-4	0	B 3.1.4-5	0
B 3.0-5	0	B 3.1.4-6	0
B 3.0-6	0	B 3.1.4-7	0
B 3.0-7	0	B 3.1.4-8	0
B 3.0-8	0	B 3.1.4-9	0
B 3.0-9	0	B 3.1.5-1	1
B 3.0-10	0	B 3.1.5-2	0
B 3.0-11	0	B 3.1.5-3	0
B 3.0-12	0	B 3.1.5-4	0
B 3.0-13	0	B 3.1.6-1	1
		B 3.1.6-2	0
		B 3.1.6-3	0
		B 3.1.6-4	0
		B 3.1.6-5	0

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.1 Bases (continued)		Chapter 3.2 Bases (continued)	
B 3.1.7-1	0	B 3.2.3-1	0
B 3.1.7-2	0	B 3.2.3-2	1
B 3.1.7-3	0	B 3.2.3-3	0
B 3.1.7-4	1	B 3.2.4-1	0
B 3.1.7-5	1	B 3.2.4-2	0
B 3.1.7-6	1	B 3.2.4-3	0
B 3.1.7-7	1	B 3.2.4-4	0
B 3.1.8-1	0	B 3.2.4-5	0
B 3.1.8-2	0	B 3.2.4-6	0
B 3.1.8-3	0	B 3.2.4-7	0
B 3.1.8-4	0		
B 3.1.8-5	0		
B 3.1.8-6	1	Chapter 3.3 Bases	
B 3.1.8-7	0		
		B 3.3.1-1	0
		B 3.3.1-2	0
Chapter 3.2 Bases		B 3.3.1-3	0
		B 3.3.1-4	0
B 3.2.1-1	0	B 3.3.1-5	0
B 3.2.1-2	0	B 3.3.1-6	0
B 3.2.1-3	0	B 3.3.1-7	0
B 3.2.1-4	0	B 3.3.1-8	1
B 3.2.1-5	0	B 3.3.1-9	0
B 3.2.1-6	0	B 3.3.1-10	1
B 3.2.1-7	0	B 3.3.1-11	1
B 3.2.1-8	0	B 3.3.1-12	1
B 3.2.1-9	0	B 3.3.1-13	0
B 3.2.1-10	0	B 3.3.1-14	0
B 3.2.1-11	0	B 3.3.1-15	1
B 3.2.2-1	0	B 3.3.1-16	0
B 3.2.2-2	0	B 3.3.1-17	0
B 3.2.2-3	0	B 3.3.1-18	0
B 3.2.2-4	0	B 3.3.1-19	0
B 3.2.2-5	0	B 3.3.1-20	0
B 3.2.2-6	0	B 3.3.1-21	0
B 3.2.2-7	0	B 3.3.1-22	0

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.3 Bases (continued)		Chapter 3.3 Bases (continued)	
B 3.3.1-23	0	B 3.3.1-59	1
B 3.3.1-24	0	B.3.3.1-60	1
B 3.3.1-25	0	B.3.3.1-61	1
B 3.3.1-26	0	B 3.3.2-1	0
B 3.3.1-27	0	B 3.3.2-2	0
B 3.3.1-28	0	B 3.3.2-3	0
B 3.3.1-29	0	B 3.3.2-4	0
B 3.3.1-30	1	B 3.3.2-5	0
B 3.3.1-31	0	B 3.3.2-6	0
B 3.3.1-32	0	B 3.3.2-7	0
B 3.3.1-33	1	B 3.3.2-8	0
B 3.3.1-34	0	B 3.3.2-9	0
B 3.3.1-35	0	B 3.3.2-10	0
B 3.3.1-36	0	B 3.3.2-11	0
B 3.3.1-37	0	B 3.3.2-12	0
B 3.3.1-38	0	B 3.3.2-13	0
B 3.3.1-39	0	B 3.3.2-14	0
B 3.3.1-40	0	B 3.3.2-15	0
B 3.3.1-41	0	B 3.3.2-16	0
B 3.3.1-42	0	B 3.3.2-17	0
B 3.3.1-43	0	B 3.3.2-18	0
B 3.3.1-44	0	B 3.3.2-19	0
B 3.3.1-45	0	B 3.3.2-20	0
B 3.3.1-46	1	B 3.3.2-21	0
B 3.3.1-47	0	B 3.3.2-22	0
B 3.3.1-48	0	B 3.3.2-23	0
B 3.3.1-49	0	B 3.3.2-24	0
B 3.3.1-50	0	B 3.3.2-25	0
B 3.3.1-51	0	B 3.3.2-26	0
B 3.3.1-52	1	B 3.3.2-27	0
B 3.3.1-53	1	B 3.3.2-28	0
B 3.3.1-54	1	B 3.3.2-29	0
B 3.3.1-55	1	B 3.3.2-30	0
B 3.3.1-56	1	B 3.3.2-31	0
B 3.3.1-57	1	B 3.3.2-32	0
B 3.3.1-58	1	B 3.3.2-33	0

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.3 Bases (continued)		Chapter 3.3 Bases (continued)	
B 3.3.2-34	0	B 3.3.3-11	0
B 3.3.2-35	0	B 3.3.3-12	0
B 3.3.2-36	0	B 3.3.3-13	0
B 3.3.2-37	0	B 3.3.3-14	0
B 3.3.2-38	0	B 3.3.3-15	0
B 3.3.2-39	0	B 3.3.3-16	1
B 3.3.2-40	0	B 3.3.3-17	1
B 3.3.2-41	0	B 3.3.4-1	0
B 3.3.2-42	0	B 3.3.4-2	0
B 3.3.2-43	0	B 3.3.4-3	1
B 3.3.2-44	0	B 3.3.4-4	0
B 3.3.2-45	0	B 3.3.4-5	0
B 3.3.2-46	0	B 3.3.4-6	1
B 3.3.2-47	0	B 3.3.5-1	0
B 3.3.2-48	0	B 3.3.5-2	0
B 3.3.2-49	0	B 3.3.5-3	0
B 3.3.2-50	0	B 3.3.5-4	0
B 3.3.2-51	1	B 3.3.5-5	0
B 3.3.2-52	1	B 3.3.5-6	0
B 3.3.2-53	1	B 3.3.5-7	0
B 3.3.2-54	1	B 3.3.5-8	0
B 3.3.2-55	0	B 3.3.6-1	1
B 3.3.2-56	0	B 3.3.6-2	1
B 3.3.2-57	0	B 3.3.6-3	1
B 3.3.2-58	0	B 3.3.6-4	1
B 3.3.2-59	0	B 3.3.6-5	1
B 3.3.3-1	0	B 3.3.6-6	1
B 3.3.3-2	0	B 3.3.6-7	0
B 3.3.3-3	0	B 3.3.6-8	0
B 3.3.3-4	0	B 3.3.7-1	1
B 3.3.3-5	0	B 3.3.7-2	1
B 3.3.3-6	0	B 3.3.7-3	1
B 3.3.3-7	0	B 3.3.7-4	1
B 3.3.3-8	0	B 3.3.7-5	1
B 3.3.3-9	0	B 3.3.7-6	1
B 3.3.3-10	0	B 3.3.7-7	1

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.3 Bases (continued)		Chapter 3.4 Bases (continued)	
B 3.3.7-8	0	B 3.4.4-3	0
B 3.3.8-1	1	B 3.4.5-1	0
B 3.3.8-2	1	B 3.4.5-2	0
B 3.3.8-3	1	B 3.4.5-3	0
B 3.3.8-4	1	B 3.4.5-4	0
B 3.3.8-5	0	B 3.4.5-5	0
B 3.3.8-6	1	B 3.4.6-1	0
B 3.3.8-7	0	B 3.4.6-2	0
B 3.3.9-1	0	B 3.4.6-3	0
B 3.3.9-2	1	B 3.4.6-4	0
B 3.3.9-3	0	B 3.4.7-1	0
B 3.3.9-4	0	B 3.4.7-2	0
B 3.3.9-5	0	B 3.4.7-3	0
B 3.3.9-6	1	B 3.4.7-4	0
B 3.3.9-7	1	B 3.4.7-5	0
		B 3.4.8-1	0
		B 3.4.8-2	0
Chapter 3.4 Bases	0	B 3.4.8-3	0
		B 3.4.9-1	0
B 3.4.1-1	0	B 3.4.9-2	0
B 3.4.1-2	0	B 3.4.9-3	0
B 3.4.1-3	0	B 3.4.9-4	0
B 3.4.1-4	0	B 3.4.10-1	0
B 3.4.1-5	0	B 3.4.10-2	0
B 3.4.2-1	0	B 3.4.10-3	0
B 3.4.2-2	0	B 3.4.10-4	0
B 3.4.2-3	0	B 3.4.11-1	0
B 3.4.3-1	0	B 3.4.11-2	0
B 3.4.3-2	0	B 3.4.11-3	0
B 3.4.3-3	0	B 3.4.11-4	0
B 3.4.3-4	0	B 3.4.11-5	0
B 3.4.3-5	0	B 3.4.11-6	0
B 3.4.3-6	0	B 3.4.11-7	0
B 3.4.3-7	0	B 3.4.11-8	0
B 3.4.4-1	0	B 3.4.12-1	0
B 3.4.4-2	0	B 3.4.12-2	0

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.4 Bases (continued)		Chapter 3.5 Bases	
B 3.4.12-3	0	B 3.5.1-1	0
B 3.4.12-4	0	B 3.5.1-2	0
B 3.4.12-5	0	B 3.5.1-3	0
B 3.4.12-6	0	B 3.5.1-4	0
B 3.4.12-7	0	B 3.5.1-5	0
B 3.4.12-8	0	B 3.5.1-6	0
B 3.4.12-9	0	B 3.5.1-7	0
B 3.4.12-10	1	B 3.5.1-8	0
B 3.4.12-11	0	B 3.5.2-1	0
B 3.4.12-12	0	B 3.5.2-2	0
B 3.4.12-13	0	B 3.5.2-3	0
B 3.4.13-1	0	B 3.5.2-4	0
B 3.4.13-2	0	B 3.5.2-5	0
B 3.4.13-3	0	B 3.5.2-6	0
B 3.4.13-4	0	B 3.5.2-7	0
B 3.4.13-5	0	B 3.5.2-8	0
B 3.4.13-6	0	B 3.5.2-9	0
B 3.4.14-1	0	B 3.5.2-10	1
B 3.4.14-2	1	B 3.5.2-11	1
B 3.4.14-3	0	B 3.5.2-12	1
B 3.4.14-4	0	B 3.5.3-1	0
B 3.4.14-5	0	B 3.5.3-2	0
B 3.4.14-6	0	B 3.5.3-3	0
B 3.4.14-7	0	B 3.5.4-1	0
B 3.4.15-1	0	B 3.5.4-2	0
B 3.4.15-2	0	B 3.5.4-3	0
B 3.4.15-3	0	B 3.5.4-4	0
B 3.4.15-4	0	B 3.5.4-5	0
B 3.4.15-5	0	B 3.5.4-6	0
B 3.4.15-6	0	B 3.5.5-1	0
B 3.4.15-7	0	B 3.5.5-2	0
B 3.4.16-1	1	B 3.5.5-3	0
B 3.4.16-2	1	B 3.5.5-4	0
B 3.4.16-3	0		
B 3.4.16-4	0		
B 3.4.16-5	0		

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.6 Bases		Chapter 3.6 Bases (continued)	
B 3.6.1-1	0	B 3.6.6-5	0
B 3.6.1-2	0	B 3.6.6-6	0
B 3.6.1-3	0	B 3.6.6-7	0
B 3.6.1-4	1	B 3.6.6-8	0
B 3.6.2-1	1	B 3.6.6-9	1
B 3.6.2-2	0	B 3.6.6-10	1
B 3.6.2-3	0	B 3.6.6-11	1
B 3.6.2-4	1	B 3.6.7-1	1
B 3.6.2-5	0	B 3.6.7-2	0
B 3.6.2-6	0	B 3.6.7-3	0
B 3.6.2-7	0	B 3.6.8-1	0
B 3.6.3-1	0	B 3.6.8-2	0
B 3.6.3-2	0	B 3.6.8-3	0
B 3.6.3-3	0	B 3.6.8-4	0
B 3.6.3-4	0	B 3.6.8-5	0
B 3.6.3-5	0		
B 3.6.3-6	0		
B 3.6.3-7	0	Chapter 3.7 Bases	
B 3.6.3-8	0		
B 3.6.3-9	0	B 3.7.1-1	0
B 3.6.3-10	0	B 3.7.1-2	1
B 3.6.3-11	0	B 3.7.1-3	1
B 3.6.3-12	0	B 3.7.1-4	1
B 3.6.3-13	0	B 3.7.1-5	1
B 3.6.3-14	0	B 3.7.2-1	0
B 3.6.4-1	0	B 3.7.2-2	0
B 3.6.4-2	0	B 3.7.2-3	0
B 3.6.4-3	0	B 3.7.2-4	0
B 3.6.5-1	0	B 3.7.2-5	0
B 3.6.5-2	0	B 3.7.2-6	0
B 3.6.5-3	0	B 3.7.3-1	0
B 3.6.5-4	0	B 3.7.3-2	1
B 3.6.6-1	0	B 3.7.3-3	1
B 3.6.6-2	0	B 3.7.3-4	1
B 3.6.6-3	0	B 3.7.3-5	1
B 3.6.6-4	0	B 3.7.4-1	1

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.7 Bases (continued)		Chapter 3.7 Bases (continued)	
B 3.7.4-2	1	B 3.7.10-6	1
B 3.7.4-3	0	B 3.7.10-7	0
B 3.7.4-4	1	B 3.7.11-1	1
B 3.7.4-5	0	B 3.7.11-2	0
B 3.7.5-1	1	B 3.7.11-3	1
B 3.7.5-2	0	B 3.7.12-1	0
B 3.7.5-3	0	B 3.7.13-1	0
B 3.7.5-4	1	B 3.7.13-2	0
B 3.7.5-5	1	B 3.7.13-3	1
B 3.7.5-6	1	B 3.7.13-4	0
B 3.7.5-7	1	B 3.7.13-5	0
B 3.7.5-8	1	B 3.7.13-6	0
B 3.7.5-9	1	B 3.7.13-7	0
B 3.7.5-10	1	B 3.7.14-1	0
B 3.7.6-1	0	B 3.7.15-1	1
B 3.7.6-2	0	B 3.7.15-2	1
B 3.7.6-3	0	B 3.7.15-3	1
B 3.7.7-1	0	B 3.7.16-1	1
B 3.7.7-2	0	B 3.7.16-2	1
B 3.7.7-3	0	B 3.7.16-3	0
B 3.7.7-4	1	B 3.7.17-1	1
B 3.7.7-5	0	B 3.7.17-2	0
B 3.7.8-1	0	B 3.7.17-3	0
B 3.7.8-2	0	B 3.7.18-1	1
B 3.7.8-3	0	B 3.7.18-2	0
B 3.7.8-4	1	B 3.7.18-3	0
B 3.7.8-5	0		
B 3.7.9-1	0		
B 3.7.9-2	0	Chapter 3.8 Bases	
B 3.7.9-3	0		
B 3.7.9-4	0	B 3.8.1-1	0
B 3.7.10-1	0	B 3.8.1-2	0
B 3.7.10-2	1	B 3.8.1-3	0
B 3.7.10-3	1	B 3.8.1-4	0
B 3.7.10-4	0	B 3.8.1-5	0
B 3.7.10-5	0	B 3.8.1-6	0

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision
Chapter 3.8 Bases (continued)		Chapter 3.8 Bases (continued)	
B 3.8.1-7	0	B 3.8.3-9	1
B 3.8.1-8	0	B 3.8.4-1	0
B 3.8.1-9	0	B 3.8.4-2	0
B 3.8.1-10	0	B 3.8.4-3	0
B 3.8.1-11	0	B 3.8.4-4	0
B 3.8.1-12	0	B 3.8.4-5	0
B 3.8.1-13	0	B 3.8.4-6	0
B 3.8.1-14	0	B 3.8.4-7	0
B 3.8.1-15	0	B 3.8.4-8	1
B 3.8.1-16	0	B 3.8.5-1	0
B 3.8.1-17	0	B 3.8.5-2	0
B 3.8.1-18	0	B 3.8.5-3	0
B 3.8.1-19	0	B 3.8.5-4	1
B 3.8.1-20	0	B 3.8.5-5	0
B 3.8.1-21	0	B 3.8.6-1	0
B 3.8.1-22	0	B 3.8.6-2	0
B 3.8.1-23	0	B 3.8.6-3	0
B 3.8.1-24	0	B 3.8.6-4	0
B 3.8.1-25	0	B 3.8.6-5	0
B 3.8.1-26	0	B 3.8.6-6	0
B 3.8.1-27	0	B 3.8.7-1	0
B 3.8.2-1	0	B 3.8.7-2	0
B 3.8.2-2	0	B 3.8.7-3	0
B 3.8.2-3	0	B 3.8.7-4	0
B 3.8.2-4	0	B 3.8.8-1	0
B 3.8.2-5	0	B 3.8.8-2	0
B 3.8.2-6	0	B 3.8.8-3	1
B 3.8.3-1	0	B 3.8.8-4	0
B 3.8.3-2	0	B 3.8.8-5	0
B 3.8.3-3	1	B 3.8.9-1	0
B 3.8.3-4	1	B 3.8.9-2	0
B 3.8.3-5	1	B 3.8.9-3	0
B 3.8.3-6	1	B 3.8.9-4	0
B 3.8.3-7	1	B 3.8.9-5	0
B 3.8.3-8	1		

LIST OF EFFECTIVE PAGES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The Chemical and Volume Control System can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 200 cal/gm average fuel pellet enthalpy at the hot spot in irradiated fuel for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are the main steam line break (MSLB) and inadvertent boron dilution accidents, as described in the FSAR (Refs. 2 and 3). In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition (automatic rod withdrawal is no longer available);

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is

(continued)

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 (automatic rod withdrawal is no longer available) 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 the measured boron concentration. Thereafter, any significant deviations

(continued)

BASES

BACKGROUND (continued)

approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 2) or RTP (automatic rod withdrawal is no longer available), loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within Technical Specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting End of Cycle (EOC) Life MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

MTC satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be 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 less negative than a given lower bound. The MTC is most positive near BOC; this upper bound must not be exceeded. This maximum upper limit occurs near BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE until the surveillance interval expires. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the 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 rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature $\geq 500^{\circ}F$ 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, Section 15.4.3.
4. FSAR, Section 4.3.1.
5. FSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability, " 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 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 four control banks and five shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Three shutdown banks (C, D, and E) consist of a single group. 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 banks are used for precise reactivity control of the reactor. The positions of the control banks are controlled manually using the Rod Control System. Automatic rod control is available for insertion only. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This

(continued)

BASES

ACTIONS (continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to MODE 3 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.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. FSAR, Chapter 15, Section 15.1.5.

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 Capability, " 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 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 four control banks and five shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Three shutdown banks (C, D, and E) consist of a single group. 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 are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are controlled manually using the Rod Control System. Automatic rod control is available for insertion only. They are capable of adding reactivity very quickly (compared to borating or diluting).

(continued)

BASES

LCO (continued)

b. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single rod, ensures high confidence that the position uncertainty of the corresponding rod group is within the assumed values used in the analysis (that specified rod group insertion limits).

These requirements ensure that rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that $F_Q(Z)$ satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided

(continued)

BASES

ACTIONS

A.1 (continued)

the nonindicating rods have not been moved. The alternate use of peaking factor and SDM verification is limited to those rodded core locations where rod position can not be determined by incore detectors. These locations are either not instrumented (or has an out of service incore thimble) or not face adjacent to instrumented assembly locations. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. While in Condition A, the performance of Required Action A.1 can be used with the Bank Demand Position Indication System (group step counters) to verify alignment limits are met for SR 3.1.4.1.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3, and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors, this will minimize the potential for rod misalignment.

The Immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording reactor coolant system T_{avg} help to assure that significant changes in power

(continued)

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods may be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that $F_Q(Z)$ satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. The alternate use of peaking factor and SDM verification is limited to those rodded core locations where rod position can not be determined by incore detectors. These locations are either not instrumented (or has an out of service incore thimble) or not face adjacent to instrumented assembly locations. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. While in Condition B, the performance of Required Action B.3 can be used with the Bank Demand Position Indication System (group step counters) to verify alignment limits are met for SR 3.1.4.1. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication (Ref. 4).

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using the movable incore detectors.

(continued)

BASES

ACTIONS (continued)

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the digital rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. While in Condition D, the performance of Required Actions D.1.1 and D.1.2 can be used with the plant computer demand position (which receives the same pulses as the group step counters) to verify alignment limits are met for SR 3.1.4.1.

D.2

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

E.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control banks and at 18, 210, and 228 steps withdrawn for the shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication. Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.7.1 (continued) This surveillance is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 13. 2. FSAR, Chapter 15. 3. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification," February 1994. 4. Amendment 61 to Callaway Plant Facility Operating License NPF-30, February 1, 1991.

BASES

LCO (continued)

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 during the performance of PHYSICS TESTS provided:

a. RCS lowest operating loop average temperature is $\geq$ 541°F;

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

c. THERMAL POWER is $\leq$ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest operating loop T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with an operating loop's temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

(continued)

BASES

ACTIONS (continued)

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The required power range and intermediate range channels must be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each OPERABLE power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest operating loop T_{avg} is $\geq$ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is $\leq$ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a manner that could invalidate the safety analysis assumptions.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidates the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for two or more OPERABLE excore channels is outside its specified limits.

RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. Axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

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 restrict the range of power distributions that are used as initial conditions in the analyses of Condition II, III, or IV events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the loss of flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

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

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks. Automatic rod control is available for insertion only.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with a random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when 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 and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. In cases where an inoperable channel is placed in the tripped condition indefinitely to satisfy the Required Action of an LCO, the logic configurations are reduced to one-out-of-two and one-out-of-three where tripping of an additional channel, for any reason, would result in a reactor trip. To allow for surveillance testing or setpoint adjustment of other channels while in this condition, several Required Actions allow the inoperable channel to be bypassed. Bypassing the inoperable channel creates a two-out-of-two or two-out-of-three logic configuration allowing a channel to be tripped for testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

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

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

1. Manual Reactor Trip (continued)

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if one or more shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal (automatic rod withdrawal is no longer available) is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods and if all rods are fully inserted. If the rods cannot be withdrawn from the core and all of the rods are fully inserted, there is no need to be able to trip the reactor. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator (SG) Water Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

a. Power Range Neutron Flux - High

The Power Range Neutron Flux - High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations and will prevent fuel melting, providing protection for the safety limit on linear heat rate. These excursions can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux - High channels to be OPERABLE (two-out-of-four trip logic). The Trip Setpoint is $\leq$ 109% RTP.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux - High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels. In these MODES, the Power Range Neutron Flux - High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux - Low

The LCO requirement for the Power Range Neutron Flux - Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux - Low channels to be OPERABLE (two-out-of-four trip logic). The Trip Setpoint is $\leq$ 25% RTP.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

b. Power Range Neutron Flux - Low (continued)

setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trip uses the same channels as discussed for Function 2 above.

Power Range Neutron Flux - High Positive Rate

The Power Range Neutron Flux - High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux - High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE (two-out-of-four trip logic). The Trip Setpoint is $\leq$ 4% RTP with a time constant $\geq$ 2 seconds.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE In MODE 3, 4, 5, or 6, the Power Range Neutron Flux- High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Power Range Neutron Flux - High Positive Rate (continued)

operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup (automatic rod withdrawal is no longer available). This trip Function provides redundant protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function (one-out-of-two trip logic). The Trip Setpoint is ≤ 25% RTP.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux - High Setpoint trip and the Power Range Neutron Flux - High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 2 below the P-6 setpoint, the Source Range Neutron Flux trip Function provides core protection for reactivity accidents. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

4. Intermediate Range Neutron Flux (continued)

the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup (automatic rod withdrawal is no longer available). This trip Function provides redundant protection to the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled manual withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. Therefore, the functional capability at the Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. This Function uses one-out-of-two trip logic. The Trip Setpoint is $\leq$ 1.0 E5 cps. The outputs of the Function to RTS logic are not required OPERABLE in MODE 6 or when all rods are fully inserted and the Rod Control System is incapable of rod withdrawal.

The Source Range Neutron Flux trip 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, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux - Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range neutron

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

6. Overtemperature ΔT (continued)

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power, either through automatic rod insertion or through operator action. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE (two-out-of-four trip logic). Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux - High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. The Overpower ΔT trip also provides protection to mitigate the consequences of small steamline breaks, as reported in Reference 11, and the decrease in feedwater temperature event (Ref. 13). It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

(continued)

BASES

APPLICABLE ANALYSES, LCO, AND APPLICABILITY

7. Overpower ΔT (continued)

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system.

ΔT_0 and T", as used in the Overpower ΔT trip, represent the 100% RTP values as measured by the plant for each loop. For the startup of a refueled core until reset to actual measured values (at 90-100% RTP), ΔT_0 is initially set at a value which is conservatively lower than the last measured 100% RTP ΔT_0 for each loop. Setting ΔT_0 and T" to the measured value of ΔT_0 and T" normalizes each loop's Overpower ΔT trip to the RCS loop conditions existing at the time of measurement, thus the trip reflects the equivalent full power conditions assumed for the OPΔT trip in the accident analyses. These differences in vessel ΔT and T_{avg} can result from several factors, two of them being measured RCS loop flows greater than Minimum Measured Flow and asymmetric power distributions between quadrants. While RCS loop flows are not expected to change, radial power redistribution between quadrants may occur resulting in small changes in loop-specific vessel ΔT and T_{avg} values. Accurate determination of the loop-specific vessel ΔT and T_{avg} values are made when performing the Incore/Excore quarterly recalibration under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions).

The time constants used in the lag compensation of measured ΔT (τ_3) and measured T_{avg} (τ_6) are set at 0 seconds. This setting corresponds to the 7300 NLL card values used for lag compensation of these signals. Safety analyses that credit Overpower ΔT for protection must account for these field adjustable lag cards as well as all other first order lag contributions (i.e., the combined RTD/thermowell response time and the scoop transport delay and thermal lag). The safety analyses use a total first order lag of less than or equal to 6 seconds.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. The actuation logic must be able to withstand an

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip - Low Fluid Oil Pressure and Turbine Trip - Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1 (two-out-of-four trip logic). The Trip Setpoint is $\leq$ 50% RTP.

In MODE 1, a turbine trip could cause a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent rod withdrawal;

- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux - Low reactor trip;

- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;

- the P-10 interlock provides one of the two inputs to the P-7 interlock; and

- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2 (two-out-of-four trip logic). The Trip Setpoint is 10% RTP.

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 Power Range Neutron Flux - Low and 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.

f. Turbine Impulse Pressure, P-13

The Turbine Impulse Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than 10% of the full power pressure. The

(continued)

BASES

ACTIONS (continued)

the Required Channels in Table 3.3.1-1 are specified on a per loop, per SG, per bus, or per train basis, then the Condition may be entered separately for each loop, SG, bus, or train.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the RTS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 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 (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to exit the Applicability from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, Condition C is entered if the Manual Reactor Trip Function has not been restored and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

(continued)

BASES

ACTIONS (continued)

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

Condition C 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:

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the RTS for these Functions. With one channel or train inoperable, the inoperable channel or 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 fully insert all rods and the Rod Control System must be rendered incapable of rod withdrawal within the next hour (e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator (MG) sets). The additional hour for the latter provides sufficient time to accomplish the action in an orderly manner. With the 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 remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

Condition C is modified by a Note stating that while this LCO is not met for Function 19, 20, or 21 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. This Note specifies an exception to LCO 3.0.4 for this MODE 5 transition and avoids placing the plant in a condition where control rods can be withdrawn or not fully inserted while the reactor trip system is degraded.

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

Condition D applies to the Power Range Neutron Flux - High trip Function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV. The CHANNEL CALIBRATIONs and COTs are 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 once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. 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 by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS power indications every 24 hours. If the calorimetric exceeds the NIS power indications by > 2% RTP, the NIS is not declared inoperable, but the excore channel gains must be adjusted consistent with calorimetric power. If the NIS power indication cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS power indications shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS power indications and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

calorimetric is > 2% RTP. The second Note clarifies that this 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. At lower power levels, calorimetric data are inaccurate. The 24-hour allowance after increasing THERMAL POWER above 15% RTP provides a reasonable time to attain a scheduled power plateau, establish the requisite conditions, perform the required calorimetric measurement, and make any required adjustments in a controlled, orderly manner and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use. Reference 10 provides additional, administratively imposed restrictions applicable to this SR.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 2%, the NIS channel is still OPERABLE, but must be readjusted. The purpose of the comparison is to check for differences that result from core power distribution changes that may have occurred since the last required adjustment or incore-excore calibration (SR 3.3.1.6).

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the Overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 2%. Note 2 clarifies that the 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. Note 2 allows power ascensions and associated testing to be conducted in a controlled and orderly manner, at conditions that provide acceptable results and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 12 hour or 4 hour limit, as applicable. These time limits are reasonable, based on operating experience, to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for the periods discussed above.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 5. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Setpoint verification is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint methodology.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.10 (continued)

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. This does not include verification of time delay relays. These are verified via response time testing per SR 3.3.1.16. See the discussion of ΔT_0 in the Applicable Safety Analyses for the Overtemperature ΔT and Overpower ΔT trip functions. Whenever an RTD is replaced in Function 6, 7, or 14.c, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The CHANNEL CALIBRATION of Function 6, Overtemperature ΔT, includes the axial flux difference penalty circuitry in the 7300 Process Protection System cabinets, but does not include the power range neutron detectors. SR 3.3.1.11 and its Notes 1 and 3 govern the performance and timing of the power range neutron detector plateau voltage verification.

Although not required for any safety function, the CHANNEL CALIBRATION of Function 10, Reactor Coolant Flow-Low, will ensure proper performance and normalization of the RCS flow indicators.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by three Notes. Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. Neutron detectors are excluded from the CHANNEL CALIBRATION because it is impractical to set up a test that demonstrates and adjusts neutron detector response to known values of the parameter (neutron flux) that the channel monitors. Note 1 applies to the source range proportional counters, intermediate range ion chambers, and power range ion chambers in the Nuclear Instrumentation System (NIS). Note 2 states that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. Detector plateau curves are obtained, evaluated, and compared to manufacturer's data for the intermediate and power range neutron detectors. The testing of the source range neutron detectors consists of obtaining integral bias curves, evaluating those curves, and comparing the curves to previous data. Note 3 states that the power and intermediate range detector plateau voltage verification is not required to be current until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 95% RTP. Equilibrium conditions are achieved

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

when the core is sufficiently stable at intended operating conditions to perform a meaningful detector plateau voltage verification. The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascension testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use. The source range integral bias curves are obtained under the conditions that apply during a plant outage.

The 18 month Frequency is based on past operating experience, which has shown these components usually pass the Surveillance when performed on the 18 month Frequency. The conditions for obtaining the source range integral bias curves and the power and intermediate range detector plateau voltages are described above. The other remaining portions of the CHANNEL CALIBRATIONS may be performed either during a plant outage or during plant operation.

SR 3.3.1.12

Not used.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every 18 months. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, the SI Input from ESFAS, and the Reactor Trip Bypass Breaker

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.14 (continued)

undervoltage trip mechanisms. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is performed every 18 months.

The Manual Reactor Trip TADOT shall independently verify the OPERABILITY of the undervoltage and shunt trip handswitch contacts for both the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip mechanism.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

analysis. Response time verification acceptance criteria are included in Reference 8. No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Reference 8. 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 until loss of stationary gripper coil voltage (at which point the rods are free to fall).

The safety analyses include the sum of the following response time components:

(a) Process delay times (e.g., scoop transport delay and thermal lag associated with the narrow range RCS RTDs used in the OTΔT, OPΔT, and SG low-low Vessel ΔT (Power-1, Power-2) functions) which are not testable;

(b) Sensing circuitry delay time from the time the trip setpoint is reached at the sensor until a reactor trip is generated by the SSPS;

(c) Any intentional time delay set into the trip circuitry (e.g., undervoltage relay time delay, NLL cards (lag, lead/lag, rate/lag) and NPL cards (PROM logic cards for trip time delay) associated with the OTΔT, OPΔT, and SG low-low level Vessel ΔT (Power-1, Power-2) trip functions, and NLL cards (lead/lag) associated with the low pressurizer pressure reactor trip function) to add margin or prevent spurious trip signals;

(d) For the undervoltage RCP trip function, back EMF delay from the time of the loss of the bus voltage until the back EMF voltage generated by the bus loads has decayed;

(e) The time delay for the reactor trip breakers to open; and

(f) The time delay for the control rod drive stationary gripper coil voltage to decay and the RCCA grippers to mechanically release making the rods free to fall (i.e., gripper release time measured during the performance of SR 3.1.4.3).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. Time constants are verified during the performance of SR 3.3.1.10. The response time may be verified by a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

series of overlapping tests, or other verification (e.g., Ref. 9 and Ref. 15), such that the entire response time is verified.

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 response times may be obtained from:

1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests); (2) inplace, onsite, or offsite (e.g. vendor) test measurements; or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. 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. Specific components identified in References 9 and 15 may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response time must be verified every 18 months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices (i.e., reactor trip breakers) is included in the verification. Testing of the final actuation devices measures the time delay for the reactor trip breakers to open. The time delay for the control rod drive stationary gripper coil voltage to decay and the RCCA grippers to mechanically release making the rods free to fall (i.e., gripper release time) is measured during the performance of SR 3.1.4.3 which verifies rod drop time from the beginning of decay of stationary gripper coil voltage. For surveillance testing performance, gripper release time is not included in the reactor trip system

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

instrumentation response time testing due to the difficulty in determining the precise point at which the rods are free to fall. SR 3.1.4.3 specifies a readily quantifiable time to use as a separation point for field measurements, i.e., "from the beginning of decay of stationary gripper coil voltage." The rod drop time measurement in SR 3.1.4.3 begins at the time the rod control power cabinet regulator board circuit for a specific rod group is grounded, causing the board to reduce the stationary gripper coil current to zero releasing the rod group. This is essentially the same time at which the reactor trip breaker's opening would interrupt current to the stationary gripper coil. The response time definition, "until loss of stationary gripper coil voltage, " is less quantifiable. However, the definition's provision for overlapping testing allows this testing approach since the total response time is determined. The safety analyses are satisfied as long as both surveillances, response time and rod drop time, are met. Some portions of the response time testing cannot be performed during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 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. Response time of the neutron flux signal portion of the channel shall be verified from detector output or input to the first electronic component in the channel.

REFERENCES

1. FSAR, Chapter 7.
2. FSAR, Chapter 15.
3. IEEE-279-1971.
4. 10 CFR 50.49.
5. Callaway OL Amendment No. 17 dated September 8, 1986.
6. Callaway Setpoint Methodology Report, SNP (UE)-565 dated May 1, 1984.
7. Callaway OL Amendment No. 43 dated April 14, 1989.

(continued)

BASES

REFERENCES (continued)

8. FSAR Section 16.3, Table 16.3-1.

9. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

10. RFR-16940A.

11. WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," Revision 1, January 1978.

12. NRC Generic Letter 85-09 dated May 23, 1985.

13. FSAR Section 15.1.1.

14. RFR - 18637A.

15. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

Table B 3.3.1-1
(Page 1 of 3)

	FUNCTION	NOMINAL TRIP SETPOINT [(a)]
1.	Manual Reactor Trip	N.A.
2.	Power Range Neutron Flux	
	a. High	$\leq$ 109% RTP
	b. Low	$\leq$ 25% RTP
3.	Power Range Neutron Flux Rate - High Positive Rate	$\leq$ 4% RTP with time constant $\geq$ 2 sec.
4.	Intermediate Range Neutron Flux	$\leq$ 25% RTP
5.	Source Range Neutron Flux	$\leq$ 1.0E5 CPS
6.	Overtemperature ΔT	See Table 3.3.1-1 Note 1.
7.	Overpower ΔT	See Table 3.3.1-1 Note 2.
8.	Pressurizer Pressure	
	a. Low	$\geq$ 1885 psig
	b. High	$\leq$ 2385 psig
9.	Pressurizer Water Level - High	$\leq$ 92% of instrument span
10.	Reactor Coolant Flow - Low	$\geq$ 90% of loop minimum measured flow (MMF=95,660 gpm)

(continued)

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K values per Reference 14.

Table B 3.3.1-1
(Page 2 of 3)

FUNCTION	NOMINAL TRIP SETPOINT [(a)]
11. Not used.	
12. Undervoltage RCPs	≥ 10,584 Vac
13. Underfrequency RCPs	≥ 57.2 Hz
14. Steam Generator (SG) Water Level Low-Low	
a. Steam Generator Water Level Low-Low (Adverse Containment Environment)	≥ 20.2% of narrow range instrument span
b. Steam Generator Water Level Low-Low (Normal Containment Environment)	≥ 14.8% of narrow range instrument span
c. Vessel ΔT Equivalent including delay timers - Trip Time Delay	
(1) Vessel ΔT (Power-1)	≤ Vessel ΔT Equivalent to 12.41% RTP (with a time delay ≤ 232 sec.)
(2) Vessel ΔT (Power- 2)	≤ Vessel ΔT Equivalent to 22.41% RTP (with a time delay ≤ 122 sec.)
d. Containment Pressure - Environmental Allowance Modifier	≤ 1.5 psig
15. Not used.	

(continued)

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K values per Reference 14.

Table B 3.3.1-1
(Page 3 of 3)

	FUNCTION	NOMINAL TRIP SETPOINT [a]
16.	Turbine Trip	
	a. Low Fluid Oil Pressure	$\geq$ 598.94 psig
	b. Turbine Stop Valve Closure	$\geq$ 1% open
17.	Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	N.A.
18.	Reactor Trip System Interlocks	
	a. Intermediate Range Neutron Flux, P-6	$\geq$ 1.0E-10 amps
	b. Low Power Reactor Trips Block, P-7	N.A.
	c. Power Range Neutron Flux, P-8	$\leq$ 48% RTP
	d. Power Range Neutron Flux, P-9	$\leq$ 50% RTP
	e. Power Range Neutron Flux, P-10	10% RTP
	f. Turbine Impulse Pressure, P-13	$\leq$ 10% Turbine Power
19.	Reactor Trip Breakers	N.A.
20.	Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	N.A.
21.	Automatic Trip Logic	N.A.

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K values per Reference 14.

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;
- Signal processing equipment including 7300 Process Protection System, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays and Balance of Plant (BOP) ESFAS circuitry: initiate the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.8 (continued)

experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of Reference 6.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

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. This does not include verification of time delay relays. These are verified via response time testing per SR 3.3.2.10.

Whenever an RTD is replaced in Function 5.e.(3) or 6.d.(3), the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

SR 3.3.2.10

This SR verifies the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response time verification acceptance criteria are included in Reference 9. No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Reference 9. Individual component response times are not modeled in the analyses. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

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 trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position). The safety analyses include the sum of the following response time components:

a. Process delay times (e.g., scoop transport delay and thermal lag associated with the narrow range RCS RTDs used in the SG low-low level Vessel ΔT (Power-1, Power-2) functions) which are not testable;

b. Sensing circuitry delay time from the time the trip setpoint is reached at the sensor until an ESFAS actuation signal is generated by the SSPS (response time testing associated with LSELS and BOP-ESFAS is discussed under SR 3.3.5.4 and SR 3.3.6.6);

c. Any intentional time delay set into the trip circuitry (e.g., NLL cards (lag) and NPL cards (PROM logic cards for trip time delay) associated with the SG low-low level Vessel ΔT (Power-1, Power-2) trip functions, NLL cards (lead/lag) associated with the steam line pressure high negative rate trip function) to add margin or prevent spurious trip signals; and

d. The time for the final actuation devices to reach the required functional state (e.g., valve stroke time, pump or fan spin-up time).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. Time constants are verified during the performance of SR 3.3.2.9. The response time may be verified by a series of overlapping tests, or other verification (e.g., Ref. 10 and Ref. 14), such that the entire response time is verified.

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 response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests); (2) inplace, onsite, or offsite (e.g. vendor) test measurements; or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response time in the overall verification of the protection system channel response time. 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. Specific components identified in References 10 and 14 may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

ESF RESPONSE TIME verification is performed on an 18 month STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices, which make up the bulk of the response time, is included in the verification of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 900 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT for the P-4 Reactor Trip Interlock. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.11 (continued)

relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The 18 month Frequency is based on operating experience. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint. This TADOT does not include the circuitry associated with steam dump operation since it is control grade circuitry.

SR 3.3.2.12

SR 3.3.2.12 is the performance of a monthly COT on ESFAS Function 6.h, "AFW Pump Suction Transfer on Suction Pressure - Low." A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

A COT is performed to ensure the channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.2-1.

The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

SR 3.3.2.13

SR 3.3.2.13 is the performance of a SLAVE RELAY TEST as described in SR 3.3.2.6, except that SR 3.3.2.13 has a Note specifying that it applies only to slave relays K602, K622, K624, K630, K740, and K741. These slave relays are tested with a Frequency of 18 months and prior to entering MODE 4 for Functions 1.b, 3.a.(2), and 7.a whenever the unit has been in MODE 5 or 6 for > 24 hours, if not performed within the previous 92 days (Reference 12). The 18 month Frequency for these slave relays is based on the need to perform this Surveillance under the conditions that apply during a unit outage to avoid the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.14

SR 3.3.2.14 is the performance of a SLAVE RELAY TEST as described in SR 3.3.2.6, except that SR 3.3.2.14 has a Note specifying that it applies only to slave relay K620. This slave relay is tested with a Frequency of 18 months and prior to entering MODE 3 for Function 5.a whenever the unit has been in MODE 5 or 6 for > 24 hours, if not performed within the previous 92 days (Reference 12). The 18 month Frequency for this slave relay is based on the need to perform this Surveillance under the conditions that apply during a unit outage to avoid the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The SLAVE RELAY TEST of relay K620 does not include the circuitry associated with the main feedwater pump trip solenoids since that circuitry serves no required safety function.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. Callaway Setpoint Methodology Report (NSSS), SNP (UE)-565 dated May 1, 1984, and Callaway Instrument Loop Uncertainty Estimates (BOP), J-U-GEN.
7. Not used.
8. Callaway OL Amendment No. 64 dated October 9, 1991.
9. FSAR Section 16.3, Table 16.3-2.
10. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
11. Callaway OL Amendment No. 43 dated April 14, 1989.
12. SLNRC 84-0038 dated February 27, 1984.
13. Callaway OL Amendment No. 117 dated October 1, 1996.
14. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

Table B 3.3.2-1
(Page 1 of 5)

FUNCTION	NOMINAL TRIP SETPOINT [(a)]
1. Safety Injection	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure - High 1	≤ 3.5 psig
d. Pressurizer Pressure - Low	≥ 1849 psig
e. Steam Line Pressure – Low	≥ 615 psig
2. Containment Spray	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure High-3	≤ 27.0 psig
3. Containment Isolation	
a. Phase A Isolation	
(1) Manual Initiation	N.A.
(2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
(3) Safety Injection	See Function 1 (Safety Injection).

(continued)

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value.

BASES

ACTIONS

G.1 (continued)

provide an alternate means for RVLIS. These three parameters provide diverse information to verify there is adequate core cooling. When Containment Radiation Level (High Range) monitors (GTRIC0059 and GTRIC0060 or GTRR0060) are inoperable, the PASS System is used as an alternate method to obtain RCS and containment atmosphere samples.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure 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. The RM-23 unit display for loop GTR-0059 and either the RM-23 unit display or the GTRR0060 recorder for loop GTR-0060 must be used to perform the CHANNEL CHECK of the Containment Radiation Level (High Range) monitors.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized. The containment hydrogen analyzers are not normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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. This SR is modified by a Note that excludes neutron detectors. Neutron detectors are excluded from the CHANNEL CALIBRATION because it is impractical to set up a test that demonstrates and adjusts neutron detector response to known values of the parameter (neutron flux) that the channel monitors. The Note applies to the Gamma-Metrics fission chambers associated with the indicators discussed in the LCO Bases. Containment Radiation Level (High Range) CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/hr and a one point calibration check of the detector below 10R/hr with an installed or portable gamma source. The Frequency is based on operating experience and consistency with the typical industry refueling cycle. During performance of the CHANNEL CALIBRATION for the Containment Radiation Level (High Range) monitors, verification of the RM-23 unit display and alarm functions is required. In addition, recorder GTRR0060 is included in the CHANNEL CALIBRATION of loop GTR-0060.

Whenever an RTD is replaced in Functions 2 or 3, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. Whenever a core exit thermocouple is replaced in Functions 14, 15, 16, or 17, the next required CHANNEL CALIBRATION of the core exit thermocouples is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

REFERENCES

1. FSAR Appendix 7A.

2. NRC Letter, "Callaway Plant, Unit 1 - Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 2," B.J. Youngblood to D.F. Schnell, dated April 10, 1985.

(continued)

BASES

REFERENCES (continued)

3. ULNRC-3023 dated May 20, 1994.
4. Callaway OL Amendment No. 103 dated October 20, 1995.
5. Regulatory Guide 1.97, Rev. 2, December 1980.
6. NUREG-0737, Supplement 1, "TMI Action Items."
7. FSAR Section 7A.3.3.
8. FSAR Section 18.2.13.
9. NUREG-0830, Callaway SER Section 22, TMI Item II.F.2 and SER Supplements 3 and 4.

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, theAuxiliary Feedwater (AFW) System and the steam generator (SG) atmospheric steam dump valves (ASDs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control at the auxiliary shutdown panel (ASP) and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the auxiliary shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the required remote shutdown controls and the following instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible:

	FUNCTION	TOTAL NO. OF CHANNELS	READOUT LOCATION
1.	Source Range Neutron Flux	2	Auxiliary Shutdown Panel
2.	Reactor Trip Breaker Position	1/RTB	Reactor Trip Switchgear
3.	Pressurizer Pressure	1	Auxiliary Shutdown Panel
4.	RCS Wide Range Pressure	2	Auxiliary Shutdown Panel

(continued)

BASES

BACKGROUND (continued)

	FUNCTION	TOTAL NO. OF CHANNELS	READOUT LOCATION
5.	RCS Hot Leg Temperature	2	Auxiliary Shutdown Panel
6.	RCS Cold Leg Temperature	4	Auxiliary Shutdown Panel
7.	SG Pressure	2/SG	Auxiliary Shutdown Panel
8.	SG Level	2/SG	Auxiliary Shutdown Panel
9.	AFW Flow Rate	4	Auxiliary Shutdown Panel
10.	Reactor Coolant Pump Breaker Position	1/pump	13.8-kV Switchgear
11.	AFW Suction Pressure	3	Auxiliary Shutdown Panel
12.	Pressurizer Level	2	Auxiliary Shutdown Panel

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 3 and GDC 19 (Ref. 1).

The Remote Shutdown System satisfies Criterion 4 of 10CFR50.36(c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the instrumentation and required ASP controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation required is listed in Table 3.3.4-1 in the accompanying LCO. The required ASP controls are described in FSAR Section 7.4.3.1.1 and are listed in FASR Table 7.4-1. The remote shutdown controls not located at the ASP are described in FSAR Section 7.4.3.1.2 and are excluded from the requirements of this LCO.

(continued)

BASES

LCO (continued)

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the AFW System and the SG ASDs; and
- RCS inventory control.

A Function of the Remote Shutdown System is OPERABLE if the required number of channels needed to support the Remote Shutdown System Function identified in Table 3.3.4-1 are OPERABLE.

The remote shutdown instruments and required ASP controls covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and controls will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or6. 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 the remote shutdown instruments and required ASP controls if control room instruments or controls become unavailable.

ACTIONS

Note 1 is included which excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function listed on Table 3.3.4-1 and for each required ASP control. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

The Notes exclude the source range neutron flux detectors and reactor trip breaker and RCP breaker position indications from the CHANNEL CALIBRATION. Neutron detectors are excluded from the CHANNEL CALIBRATION because it is impractical to set up a test that demonstrates and adjusts neutron detector response to known values of the parameter (neutron flux) that the channel monitors. Depending on which source range channel is used to satisfy the LCO, Note 1 applies to the source range proportional counter in the Nuclear Instrumentation System (NIS) associated with indicator SENI0031C or to the Gamma-Metrics fission chamber associated with indicator SENI0061X. As discussed in the Bases for SR 3.3.1.11, the CHANNEL CALIBRATION of the Westinghouse NIS source range channel consists of obtaining on integral bias curve, evaluating that curve, and comparing it to previous data.

Whenever an RTD is replaced in Function 5 or 6, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 3 and GDC 19.

2. Callaway OL Amendments No. 45 dated May 16, 1989 and 108 dated March 11, 1996.

B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. If a loss of voltage condition occurs at the 4.16 kV ESF buses, undervoltage protection provided by the load shedder and emergency load sequencer (LSELS) will (Ref. 1):

a) Trip the 4.16 kV preferred normal and alternate bus feeder breakers to remove the deficient power source to protect the Class 1E equipment from damage;

b) Shed all loads from the bus except the Class 1E 480 Vac load centers and centrifugal charging pumps to prepare the buses for re-energization; and

c) Generate an LOP DG start signal.

There are two sets of undervoltage protection circuits, one for each 4.16 kV NB system bus. Each set consists of a loss of voltage and degraded voltage Function. Four potential transformers on each bus provide the necessary input voltages to the protective devices used to perform these functions. The undervoltage protection circuits are described in FSAR Section 8.3.1.1.3 (Ref. 1).

Four instantaneous undervoltage relays with an associated time delay are provided for each 4.16 kV Class 1E system bus for detecting a loss of bus voltage. The outputs are combined in a two-out-of-four logic to generate an LOP signal if the voltage is below approximately 70% for 1 second (nominal delay). The time delay prevents undesirable trips arising from transient undervoltage conditions.

Four degraded voltage bistables with associated time delays are provided for each 4.16 kV Class 1E system bus for detecting a sustained degraded voltage condition. Once the bistable has actuated, a timer in the LSELS circuitry provides an 8 second time delay to avoid false actuation on large motor starts other than an RCP. There are four of these 8-second timers per bus, one for each degraded voltage channel. The bistable outputs are then combined in a two-out-of-four logic to generate a degraded voltage signal if the voltage is below approximately 90%. Once the two-out-of-four logic is satisfied, contacts in the bus feeder breaker trip circuits close to arm the tripping circuitry. If a safety injection signal (SIS)

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 8.3.1.1.3.
2. FSAR, Chapter 15.
3. Callaway OL Amendment No. 74 dated December 16, 1992.
4. Callaway OL Amendment No. 99 dated April 18, 1995.
5. FSAR Table 16.3-2.
6. NRC Branch Technical Position PSB-1.

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge Isolation Instrumentation

BASES

BACKGROUND

Containment purge isolation instrumentation closes the containment isolation valves in the Mini-purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown.

Containment purge isolation initiates on an automatic or manual safety injection (SI) signal through the Containment Isolation - Phase A Function, or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Two gaseous radiation monitoring channels are also provided as input to the containment purge isolation. The two channels measure gaseous radiation in a sample of the containment purge exhaust. Since the purge exhaust monitors constitute a sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from either of the two radiation monitoring channels initiates containment purge isolation, which closes both inner and outer containment isolation valves in the Mini-purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge isolation gaseous radiation channels act as backup to the Phase A isolation signal to ensure closing of the purge supply and exhaust valves. They are also the means for automatically isolating containment in the event of a fuel handling accident during shutdown; however, the dose calculations performed in support of Reference 5 do not assume automatic isolation (see also the Bases for LCO 3.9.4, "Containment Penetrations"). Containment isolation in turn ensures

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment purge isolation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two push buttons in the control room.

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 actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of containment purge isolation.

Automatic Actuation Logic and Actuation Relays (BOP ESFAS) consist of the same features and operate in the same manner as described for ESFAS Function 6.c, Auxiliary Feedwater.

3. Containment Purge Exhaust Radiation - Gaseous

The LCO specifies two required Containment Purge Exhaust Radiation – Gaseous channels (GTRE0022 and GTRE0033) to ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge Isolation remains OPERABLE.

(continued)

BASES

LCO

3. Containment Purge Exhaust Radiation (continued)

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY also requires correct valve lineups and sample pump operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

4. Containment Isolation - Phase A

Containment Purge Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Containment Isolation - Phase A. Therefore, the requirements are not repeated in Table 3.3.6-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), Containment Isolation - Phase A, and Containment Purge Exhaust Radiation - Gaseous Functions are required OPERABLE in MODES 1, 2, 3, and 4. The Containment Purge Exhaust Radiation - Gaseous, Manual Initiation, and BOP ESFAS Logic Functions are also required OPERABLE during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without CORE ALTERATIONS or irradiated fuel movement within containment in progress, the containment 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 the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the measured Trip Setpoint is less conservative

(continued)

BASES

ACTIONS (continued)

than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/ train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment purge isolation gaseous radiation monitor channel. Since two containment purge exhaust gaseous radiation monitor channels are required to meet single failure criteria, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channel will respond.

B.1

Condition B applies to all Containment Purge Isolation Functions and addresses the train orientation of the BOP ESFAS actuation logic and actuation relays for these Functions. It also addresses the failure of both gaseous radiation monitoring channels, or the inability to restore a single failed gaseous radiation monitoring channel to OPERABLE status in the time allowed for Required Action A.1.

If one or more trains or manual initiation channels are inoperable, both gaseous radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust valves in their closed position is met.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to the Manual Initiation, Automatic BOP ESFAS Actuation Logic and Actuation Relays, and Containment Purge Exhaust Radiation - Gaseous Functions and addresses the train orientation of the BOP ESFAS. It also addresses the failure of both gaseous radiation monitoring channels, or the inability to restore a single failed gaseous radiation monitoring channel to OPERABLE status in the time allowed for Required Action A.1. If one or more BOP ESFAS logic trains or manual initiation channels are inoperable, both gaseous radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check. This test is required every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.6.3

A COT is performed every 92 days on each required containment purge exhaust gaseous radiation monitor channel to ensure the channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment purge system isolation. The setpoint shall be left within the two-sided calibration tolerance band on either side of the nominal value.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

SR 3.3.6.4 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 18 months. Each Manual Initiation channel is tested through the BOP ESFAS logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The channels tested have no setpoints associated with them.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.6.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.6

SR 3.3.6.6 is the performance of the required response time verification every 18 months on a STAGGERED TEST BASIS on those functions with time limits provided in Reference 3. Each verification shall include at least one train such that both trains are verified at least once per 36 months.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
2. NUREG-1366, July 22, 1993.
3. FSAR Table 16.3-2.
4. Callaway OL Amendment No. 20 dated April 10, 1987.
5. Callaway OL Amendment No. 114 dated July 15, 1996.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation

BASES

BACKGROUND	The CREVS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Control Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREVS initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)."
	The actuation instrumentation consists of two gaseous radiation channels in the control room air intake. A high radiation signal from either of these channels will initiate both trains of the CREVS. Since the radiation monitors include an air sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY. The control room operator can also initiate CREVS trains by manual switches in the control room. The CREVS is also actuated by a Phase A Isolation signal, a Fuel Building Ventilation Isolation signal (FBVIS), or a high radiation signal from the containment purge exhaust gaseous radiation channels. The Phase A Isolation Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."
APPLICABLE SAFETY ANALYSES	The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.
	The CREVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.
	In MODES 1, 2, 3, and 4, the gaseous channel actuation of the CREVS is a backup for the Phase A Isolation signal actuation. This ensures initiation of the CREVS during a loss of coolant accident or steam generator tube rupture.
	The gaseous radiation channel actuation of the CREVS in MODES 5 and 6 and during movement of irradiated fuel assemblies is the primary

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

means to ensure control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident. The probability of a waste gas decay tank rupture accident occurring during the period of time outside the Applicability (i.e., not in MODES 1-6 and with no movement of irradiated fuel assemblies occurring) is insignificant. There are no safety analyses that take credit for CREVS actuation upon an FBVIS or high containment purge exhaust radiation.

The CREVS actuation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREVS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the CREVS at any time by using either of two push buttons in the control room.

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 actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of control room ventilation isolation.

Automatic Actuation Logic and Actuation Relays (BOP ESFAS) consist of the same features and operate in the same manner as described for ESFAS Function 6.c, Auxiliary Feedwater.

3. Control Room Radiation – Control Room Air Intake

The LCO specifies two required Control Room Radiation Monitor – Control Room Air Intake gaseous channels (GKRE0004 and GKRE0005) to ensure that the radiation monitoring

(continued)

BASES

LCO

3. Control Room Radiation – Control Room Air Intake (continued)

instrumentation necessary to initiate the CREVS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY also requires correct valve lineups and sample pump operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses. The required radiation monitors' OPERABILITY is not dependent on forced flow in the control room supply duct. GKRE0004 and GKRE0005 OPERABILITY is not dependent on the status of GKHZ0013D/0057A/0150/0151, SGK02, or CGK01A and B. GKRE0004 and GKRE0005 may be considered OPERABLE with CREVS in the CRVIS mode of operation.

4. Containment Isolation - Phase A

Control Room Ventilation Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Containment Isolation - Phase A. Therefore, the requirements are not repeated in Table 3.3.7-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

All CREVS Functions, including actuation on the Containment Isolation - Phase A Function, must be OPERABLE in MODES 1, 2, 3, and4. The Manual Initiation, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), and Control Room Radiation – Control Room Air Intake Functions are also required OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. These Functions must be OPERABLE in MODES 5 and 6 for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. It the measured Trip Setpoint is less conservative than the tolerance specified by the calibration

(continued)

BASES

ACTIONS (continued)

procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

Placing a CREVS train(s) in the CRVIS mode of operation isolates the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through HEPA filters and charcoal adsorbers. This mode of operation also initiates pressurization and filtered ventilation of the air supply to the control room. Further discussion of the CRVIS mode of operation may be found in the Bases for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)," and in Reference 1.

A.1

Condition A applies to all CREVS Functions (i.e., the actuation logic train Function of the BOP ESFAS, the gaseous radiation monitor channel Function, and the manual initiation channel Function).

If one channel or train is inoperable, or one gaseous radiation monitor channel 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 CREVS train must be placed in the Control Room Ventilation Isolation Signal (CRVIS) mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two CREVS actuation logic trains (BOP ESFAS) or two manual initiation channels. Condition B is modified by a Note stating this Condition is not applicable to Function 3. Function 3 in Table 3.3.7-1 applies to the Control Room Radiation - Control Room Air Intake gaseous channels. The first Required Action is to place one CREVS train in the CRVIS mode of operation immediately.

(continued)

BASES

ACTIONS

B.1.1, B.1.2, and B.2 (continued)

This accomplishes the actuation instrumentation Function that has 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 immediately for one CREVS 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, both trains may be placed in the CRVIS mode immediately. This ensures the CREVS function is performed even in the presence of a single failure.

C.1.1, C.1.2, and C.2

Condition C applies to the failure of both gaseous radiation monitoring channels. The first Required Action is to enter the applicable Conditions and Required Actions of LCO 3.7.10 immediately for one CREVS 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. One CREVS train must also be placed in the CRVIS mode of operation within 1 hour. This accomplishes the actuation instrumentation Function that has been lost and places the unit in a conservative mode of operation. The 1 hour Completion Time allows for activities such as changing sample filters on the OPERABLE channel while in Condition A, which requires entry into Condition C.

Alternatively, both trains may be placed in the CRVIS mode within 1 hour. This ensures the CREVS function is performed even in the presence of a single failure.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Conditions A, B, or C 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.

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies when the Required Action and associated Completion Time for Conditions A, B, or C have not been met in MODE 5 or 6, or when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREVS actuation. This does not preclude movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

Either the RM-11 or RM-23 displays may be used to perform the CHANNEL CHECK for the Control Room Radiation - Control Room Air Intake gaseous channels (GKRE0004 and GKRE0005).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2

A COT is performed once every 92 days on each required control room air intake gaseous radiation monitor channel to ensure the channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CREVS actuation. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The setpoints shall be left within the two-sided calibration tolerance band on either side of the nominal value. The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.7.3

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check. This test is required every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.7.4

SR 3.3.7.4 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 18 months. Each Manual Initiation channel is tested through the BOP ESFAS logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.4 (continued)

operating experience. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The channels tested have no setpoints associated with them.

SR 3.3.7.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. FSAR Section 7.3.4 and Table 7.3-8.

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Exhaust System (EES) Actuation Instrumentation

BASES

BACKGROUND	The EES ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Emergency Exhaust System." The system initiates filtered exhaust from the fuel building following receipt of a fuel building ventilation isolation signal (FBVIS), initiated manually or automatically upon a high radiation signal (gaseous). High gaseous radiation, monitored by two channels, provides an FBVIS. Both EES trains are initiated by high radiation detected by either channel. Each channel contains a gaseous monitor. High radiation detected by either monitor initiates fuel building isolation and starts the EES. These actions function to prevent exfiltration of contaminated air by initiating filtered exhaust, which imposes a negative pressure on the fuel building. Since the radiation monitors include an air sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY. In the FBVIS mode, each train is capable of maintaining the fuel building at a negative pressure of less than or equal to 0.25 inches water gauge relative to the outside atmosphere. The EES is also actuated in the LOCA (SIS) mode as described in the Bases for LCO 3.3.2, "ESFAS Instrumentation."
APPLICABLE SAFETY ANALYSES	The EES ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1). The EES actuation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	The LCO requirements ensure that instrumentation necessary to initiate the EES is OPERABLE.

(continued)

BASES

LCO (continued)

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the EES at any time by using either of two push buttons in the control room.

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 actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation. This consists of the same features and operates in the same manner as described for ESFAS Function 6.c, Auxiliary Feedwater.

3. Fuel Building Exhaust Radiation - Gaseous

The LCO specifies two required Fuel Building Exhaust Radiation–Gaseous channels (GGRE0027 and GGRE0028) to ensure that the radiation monitoring instrumentation necessary to initiate the FBVIS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY also requires correct valve lineups, sample pump operation, and detector OPERABILITY, since these supporting features are necessary for actuation to occur under the conditions assumed by the safety analyses. The required radiation monitors remain OPERABLE if one or both Emergency Exhaust System trains are inoperable or following a Fuel Building Ventilation Isolation Signal (FBVIS). Both required radiation monitors remain OPERABLE if the normal Fuel Building exhaust flow is isolated.

The submersion dose rate basis for the nominal Trip Setpoint is specified for the gaseous monitors in the LCO. The nominal Trip Setpoint accounts for instrument uncertainties.

(continued)

BASES (continued)

APPLICABILITY

The manual and automatic EES initiation must be OPERABLE when moving irradiated fuel assemblies in the fuel building to ensure the EES operates to remove fission products associated with a fuel handling accident.

High radiation initiation of the FBVIS must be OPERABLE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the EES when the potential for a fuel handling accident exists.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the measured Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

Placing a EES train(s) in the FBVIS mode of operation isolates normal air discharge from the fuel building and initiates filtered exhaust, imposing a negative pressure on the fuel building. Further discussion of the FBVIS mode of operation may be found in the Bases for LCO 3.7.13, "Emergency Exhaust System (EES)," and in Reference 2.

A.1

Condition A applies to the actuation logic train Function of the BOP ESFAS, the gaseous radiation monitor channel Function, and the manual initiation channel Function. Condition A applies to the failure of a single actuation logic train, gaseous radiation monitor channel, or manual initiation channel. If one channel or train is inoperable, or one gaseous radiation monitor channel is inoperable, a period of 7 days is allowed to

(continued)

BASES

ACTIONS

A.1 (continued)

restore it to OPERABLE status. If the channel or train cannot be restored to OPERABLE status, one EES train must be placed in the FBVIS mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation. 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 time is the same as that provided in LCO 3.7.13.

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two EES actuation logic trains (BOP ESFAS) or two manual initiation channels. Condition B is modified by a Note stating this Condition is not applicable to Function 3. Function 3 in Table 3.3.8-1 covers the Fuel Building Exhaust Radiation – Gaseous channels. The first Required Action is to place one EES train in the FBVIS mode of operation immediately. This accomplishes the actuation instrumentation Function that has been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.13 must also be entered immediately for one EES train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.13.

Alternatively, both trains may be placed in the FBVIS mode immediately. This ensures the EES function is performed even in the presence of a single failure.

C.1.1, C.1.2, and C.2

Condition C applies to the failure of both gaseous radiation monitoring channels. The first Required Action is to enter the applicable Conditions and Required Actions of LCO 3.7.13 immediately for one EES train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.13. One EES train must also be placed in the FBVIS mode of operation within 1 hour. This accomplishes the actuation instrumentation Function that has been lost and places the unit in a conservative mode of operation. The 1 hour Completion Time allows for activities such as changing sample filters on the OPERABLE channel while in Condition A, which requires entry into Condition C.

(continued)

BASES

ACTIONS

C.1.1, C.1.2, and C.2 (continued)

Alternatively, both trains may be placed in the FBVIS mode within 1 hour. This ensures the EES function is performed even in the presence of a single failure.

D.1

Condition D applies when the Required Action and associated Completion Time for Conditions A, B, or C have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require EES actuation. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which EES Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

Either the RM-11 or RM-23 displays may be used to perform the CHANNEL CHECK for the Fuel Building Exhaust Radiation – Gaseous channels (GGRE0027 and GGRE0028).

SR 3.3.8.2

A COT is performed once every 92 days on each required fuel building exhaust gaseous radiation monitor channel to ensure the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This test verifies the capability of the instrumentation to provide the EES actuation. The setpoints shall be left within the two-sided calibration tolerance band on either side of the nominal value. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

SR 3.3.8.3 is the performance of an ACTUATION LOGIC TEST. The actuation logic is tested every 31 days on a STAGGERED TEST BASIS. All possible logic combinations are tested for each protection function. The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience. The SR is modified by a Note stating that the continuity check may be excluded. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check.

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 18 months. Each

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.4 (continued)

Manual Initiation channel is tested through the BOP ESFAS logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The channels tested have no setpoints associated with them.

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. 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. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

2. FSAR Section 7.3.3 and Table 7.3-5.

B 3.3 INSTRUMENTATION

B 3.3.9 Boron Dilution Mitigation System (BDMS)

BASES

BACKGROUND

The primary purpose of the BDMS is to mitigate the consequences of the inadvertent addition of unborated primary grade water into the Reactor Coolant System (RCS) when the plant is in MODES 2 (below P-6 setpoint), 3, 4, and 5.

The BDMS utilizes two channels of source range instrumentation. Each source range channel provides a signal to its microprocessor, which continuously records the counts per minute. At the end of each discrete one-minute interval, an algorithm compares the average counts per minute value (flux rate) of that 1 minute interval with the average counts per minute value for the previous nine, 1 minute intervals. If the flux rate during a 1 minute interval is greater than or equal to 1.7 times the flux rate during any of the prior nine 1 minute intervals, the BDMS provides a signal to initiate mitigating actions.

Upon detection of a flux multiplication by either source range instrumentation train, an alarm is sounded to alert the operator and valve movement is automatically initiated to terminate the dilution and start boration. Valves that isolate the refueling water storage tank (RWST) are opened to supply borated water to the suction of the centrifugal charging pumps, and valves which isolate the Volume Control Tank are closed to terminate the dilution.

APPLICABLE SAFETY ANALYSES

The BDMS senses abnormal increases in source range counts per minute (flux rate) and actuates VCT and RWST valves to mitigate the consequences of an inadvertent boron dilution event as described in Reference 1. The accident analyses rely on automatic BDMS actuation to mitigate the consequences of inadvertent boron dilution events. The operation of one RCS loop in MODES 2 (below P-6 setpoint), 3, 4, and5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the BDMS. With no reactor coolant loop in operation in the above MODES, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The event is successfully terminated after the volume of water from the normally closed RWST suction isolation valves to the RCS via the normal charging flow path is purged and inadvertent criticality is avoided. The primary success path for mitigation is fulfilled when the VCT suction path is isolated; however, the analysis also accounts for the volume of CVCS piping from the RWST to the RCS that must be purged since its boron content is dependent on time in cycle life and may itself represent a dilution source.

The BDMS satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

LCO 3.3.9 provides the requirements for OPERABILITY of the instrumentation that provides control room indication of core neutron levels, and that mitigates the consequences of a boron dilution event. Two redundant trains are required to be OPERABLE to provide protection against single failure. In addition, LCO 3.3.9 requires that one RCS loop shall be in operation.

Because the BDMS utilizes the source range instrumentation in its detection system, the OPERABILITY of that portion of the detection system is also part of the OPERABILITY of the Reactor Trip System. The flux multiplication algorithm, the alarms, and signals to the motor control centers for the suction valves all must be OPERABLE for a train in the system to be considered OPERABLE. As required for this LCO, the BDMS extends to, and includes, the RWST suction isolation valves (BNLCV0112D, E) and the VCT suction isolation valves (BGLCV0112B, C).

With insufficient RCS mixing volume, i.e. no RCS loop in operation, Condition C must be entered.

APPLICABILITY

The BDMS must be OPERABLE in MODES 2 (below P-6 setpoint), 3, 4, and 5 because the safety analysis identifies this system as the primary means to mitigate an inadvertent boron dilution of the RCS.

The BDMS OPERABILITY requirements are not applicable in MODES 1 and 2 (above P-6 setpoint) because an inadvertent boron dilution would be terminated by Overtemperature ΔT or operator action. The Overtemperature ΔT trip Function is discussed in LCO 3.3.1, "RTS Instrumentation."

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The BDMS trains are subject to a CHANNEL CHECK, valve closure in MODE 5, COT, CHANNEL CALIBRATION, and Response Time Testing. In addition, the requirement to verify one RCS loop in operation is subject to periodic surveillance.

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of source range 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 by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.9.2

SR 3.3.9.2 requires that valve BGV0178 be secured and closed prior to entry into MODE 5. Specification 3.9.2 requires that this valve also be secured and closed in MODE 6. Closing BGV0178 satisfies the boron dilution accident analysis assumption that flow orifice BGFO0010 limits the dilution flow rate to no more than 150 gpm in MODE 5. This Surveillance demonstrates that the valve is closed through a system walkdown. SR 3.3.9.2 is modified by a Note stating that it is only required to be performed in MODE 5. This Note requires that the surveillance be performed prior to entry into MODE 5 and every 31 days while in MODE 5. The 31 day frequency is based on engineering judgment and is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.2 (continued)

considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

SR 3.3.9.3

SR 3.3.9.3 requires the performance of a COT every 92 days, to ensure that each train of the BDMS and associated trip setpoints are fully operational. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This test shall include verification that the boron dilution flux multiplication setpoint is equal to or less than an increase of 1.7 times the count rate within a 10 minute period. The 1.7 flux multiplication setpoint is a nominal value. SR 3.3.9.3 is met if the measured setpoint is within a two-sided calibration tolerance band on either side of the nominal value. SR 3.3.9.3 is modifiedby a Note that provides a 4 hour delay in the requirement to perform this Surveillance after reducing power below the P-6 interlock. This Note allows a delay in the performance of the COT to reflect the delay allowed for the source range channels. If the plant is to remain below the P-6 setpoint for more than 4 hours, this Surveillance must be performed prior to 4 hours after reducing power below the P-6 setpoint. The Frequency of 92 days is consistent with the requirements for source range channels in Reference 2.

SR 3.3.9.4

SR 3.3.9.4 is the performance of a CHANNEL CALIBRATION every 18 months. 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. The SR is modified by a Note that neutron detectors are excluded from the CHANNEL CALIBRATION. Neutron detectors are excluded from the CHANNEL CALIBRATION because it is impractical to set up a test that demonstrates and adjusts neutron detector response to known values of the parameter (neutron flux) that the channel monitors. The Note applies to the source range proportional counters in the Nuclear Instrumentation System (NIS).

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.4 (continued)

The testing of the source range neutron detectors consists of obtaining integral bias curves, evaluating those curves, and comparing the curves previous data. The 18 month Frequency is based on operating experience and on the need to obtain integral bias curves under the conditions that apply during a plant outage. The other remaining portions of the CHANNEL CALIBRATION may be performed either during a plant outage or during plant operation.

SR 3.3.9.5

SR 3.3.9.5 is the performance of a response time test every 18 months to verify that, on a simulated or actual boron dilution flux multiplication signal, the centrifugal charging pump suction valves from the RWST open and the CVCS volume control tank discharge valves close in the required time of $\leq$ 30 seconds to reflect the analysis requirements of Reference 1.

The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

SR 3.3.9.6

SR 3.3.9.6 requires verification every 12 hours that one RCS loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing adequate mixing. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

REFERENCES

1. FSAR, Section 15.4.6.
2. Callaway OL Amendment No. 17 dated September 8, 1986.

BASES

ACTIONS (continued)

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or

c. The COMS is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized $\geq$ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero safety injection pumps and a maximum of one centrifugal charging pump are verified to be capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed with power removed from the valve operators (Refs. 10 and 11). Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 3 to the LCO.

The safety injection pumps and one centrifugal 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 cold overpressure protection control may be employed using at least two independent means to render a pump incapable of injecting into the RCS such that a single failure or single action will not result in an injection into the RCS. This may be accomplished by placing the pump control switch in pull to lock and closing at least one valve in the discharge flow path, or by closing at least

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

one valve in the discharge flow path and removing power from the valve operator, or by closing at least one manual valve in the discharge flow path under administrative controls.

The Frequency of 12 hours is sufficient, considering administrative controls and other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.4

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. 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 every 72 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.5

The RCS vent of $\geq$ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent pathway that is not locked, sealed, or otherwise secured in the open position, or

b. Once every 31 days for a valve that is locked, sealed, or otherwise secured in the open position. A removed pressurizer safety valve or open manway also fits this category.

Any passive vent path arrangement must only be open when required to be OPERABLE. This Surveillance is required if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12d.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

(continued)

BASES

BACKGROUND (continued)

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed below :

VALVE NUMBER	VALVE SIZE (in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE (gpm)
BB8948A	10	RCS Loop 1 Cold Leg SI Accu Chck	5.0
BB8948B	10	RCS Loop 2 Cold Leg SI Accu Chck	5.0
BB8948C	10	RCS Loop 3 Cold Leg SI Accu Chck	5.0
BB8948D	10	RCS Loop 4 Cold Leg SI Accu Chck	5.0
BB8949A	6	RCS Loop 1 Hot Leg SI/RHR Pump Chck	3.0
BB8949B	6	RCS Loop 2 Hot Leg SI/RHR Pump Chck	3.0
BB8949C	6	RCS Loop 3 Hot Leg SI/RHR Pump Chck	3.0
BB8949D	6	RCS Loop 4 Hot Leg SI/RHR Pump Chck	3.0
BBV0001	1.5	RCS Loop 1 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0022	1.5	RCS Loop 2 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0040	1.5	RCS Loop 3 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0059	1.5	RCS Loop 4 Cold Leg SI/Boron Injection Header Chck	0.75
BBPV8702A	12	RCS Loop 1 Hot Leg to RHR Pumps ISO	5.0
BBPV8702B	12	RCS Loop 4 Hot Leg to RHR Pumps ISO	5.0
EJ8841A	6	RHR TRNS SIS Hot Leg Loop 2 Recirc	3.0
EJ8841B	6	RHR TRNS SIS Hot Leg Loop 3 Recirc	3.0
EJHV8701A	12	RHR Pump A Suction ISO	5.0
EJHV8701B	12	RHR Pump B Suction ISO	5.0
EMV0001	2	SI Pump A Disch to Hot Leg Loop 2 Chck	1.0
EMV0002	2	SI Pump A Disch to Hot Leg Loop 3 Chck	1.0
EMV0003	2	SI Pump B Disch to Hot Leg Loop 1 Chck	1.0
EMV0004	2	SI Pump B Disch to Hot Leg Loop 4 Chck	1.0
EM8815	3	Boron Injection Header CVCS Out Check	1.5
EPV0010	2	SI Pumps to RCS Cold Leg Loop 1 Chck	1.0
EPV0020	2	SI Pumps to RCS Cold Leg Loop 2 Chck	1.0
EPV0030	2	SI Pumps to RCS Cold Leg Loop 3 Chck	1.0
EPV0040	2	SI Pumps to RCS Cold Leg Loop 4 Chck	1.0
EP8818A	6	RHR Pumps to RCS Cold Leg Loop 1 Chck	3.0
EP8818B	6	RHR Pumps to RCS Cold Leg Loop 2 Chck	3.0
EP8818C	6	RHR Pumps to RCS Cold Leg Loop 3 Chck	3.0

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the initial specific activity of the reactor coolant is greater than the LCO limit and assumes an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the initial specific activity of the secondary coolant is greater than the limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is performed for two cases of reactor coolant specific activity. Case 1 assumes a concurrent large iodine spike

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

that increases the rate of iodine release into the reactor coolant by a factor of about 500 immediately after the accident. Case 2 assumes the initial reactor coolant iodine activity is a factor of 60 higher than Case 1 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/ $\bar{E}$ μCi/gm for gross specific activity. These assumptions are discussed further in Table 15.6-4 of Reference 2.

The analysis also assumes a loss of offsite power at the same time as the reactor trip after an SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric steam dump valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within the SRP 15.6.3 fractions of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking cases.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

b. The hand control switch for SI pump A (or SI pump B) is placed in pull to lock.

Closure of EMHV8821A or EMHV8821B isolates the associated SI pump from its cold leg injection path rendering that train inoperable; however, the opposite train is prevented from exceeding runout flow conditions which would occur if the opposite pump were connected to both cold leg and hot leg injection paths. The inoperable train's pump is then placed in pull to lock to prevent unanalyzed hot leg injection via its associated 8802 valve. Although one SI train would be rendered inoperable, more than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train would be available, and the plant would be in CONDITION A.1 with a 72 hour restoration time rather than entering LCO 3.0.3.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves and relief valves. Additionally, vent and drain valves are not within the scope of this SR.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operations, and ensures correct valve positions.

SR 3.5.2.3

The ECCS pumps are normally in a standby, non-operating mode. As such, flow path piping has the potential to develop voids and pockets of

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting RHR and SI pump casings and accessible ECCS discharge piping high point vents ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. The design of the centrifugal charging pump is such that significant noncondensible gases do not collect in the pump. Therefore, it is unnecessary to require periodic pump casing venting to ensure the centrifugal charging pumps will remain OPERABLE. Accessible high point vents are those that can be reached without hazard or high radiation dose to personnel. This will also prevent water hammer, pumpcavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. The ECCS pumps are required to develop the following differential pressures on recirculation flow: 1) centrifugal charging pumps $\geq$ 2400 psid; 2) safety injection pumps $\geq$ 1445 psid; and 3) RHR pumps $\geq$ 165 psid. 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 applicable portions of the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signalor on an actual or simulated RWST Level Low-Low 1 Automatic Transfer signal coincident with an SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. The containment recirculation sump to RHR pump isolation valves (EJHV8811A/B) automatically open upon

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5 and SR 3.5.2.6 (continued)

receipt of an actual or simulated RWST Level Low-Low-1 Automatic Transfer signal coincident with an SI signal. In addition to testing that automatic function, SR 3.5.2.5 demonstrates thatthe RWST to RHR pump suction isolation valves (BNHV8812A/B) are capable of automatic closure after the EJHV8811A/B valves are fully open. The valve interlock functions are depicted in Reference 10. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

The correct position of throttle valves in the flow path is necessary for proper ECCS performance. These valves have mechanical stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6. The ECCS throttle valves are set to ensure proper flow resistance and pressure drop in the piping to each injection point in the event of a LOCA. Once set, these throttle valves are secured with locking devices and mechanical position stops. These devices help to ensure that the following safety analyses assumptions remain valid: (1) both the maximum and minimum total system resistance; (2) both the maximum and minimum branch injection line resistance; and (3) the maximum and minimum ranges of potential pump performance. These resistances and pump performance ranges are used to calculate the maximum and minimum ECCS flows assumed in the LOCA analyses of Reference 3.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.8 (continued)

Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Sections 6.3 and 15.6.
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. RFR-14801A.
8. ULNRC-2535 dated 12-18-91.
9. OL Amendment No. 68 dated 3-24-92.
10. FSAR Figure 7.6-3.

BASES

LCO (continued)

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and containment shutdown and mini-purge valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. These leakage rates are specified in the Containment Leakage Rate Testing Program. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage and < 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with Reference 4.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 15.
3. FSAR, Section 6.2.
4. FSAR Chapter 16.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

The personnel air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end. On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide local indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

The DBA that result in a release of radioactive material within containment is a loss of coolant accident (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.2% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, (Ref. 1) L_a as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a = 48.1 psig following a design basis LOCA. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	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 10CFR 50.36(c)(2)(ii).
LCO	Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.
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 3.9.4, "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 primary 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 may not be intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable

(continued)

BASES

ACTIONS (continued)

due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5 and SR 3.6.6.6

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 actuation of a containment High-3 pressure signal.

This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. Upon actuation the fans start in slow speed or, if operating, shift to slow speed and the cooling flow rate increases to the minimum value required per Figure B 3.6.6-1 to maintain cooler heat removal capacity in the "Acceptable Region". In no case shall the flow rate be less than 2000 gpm per cooler train. The value of containment cooler heat removal capacity per train is established periodically by Engineering. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

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 provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, GDC 43 and GDC 50.

(continued)

BASES

REFERENCES (continued)	2.	10 CFR 50, Appendix K.
	3.	FSAR, Section 6.2.1.
	4.	FSAR, Section 6.2.2.
	5.	ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

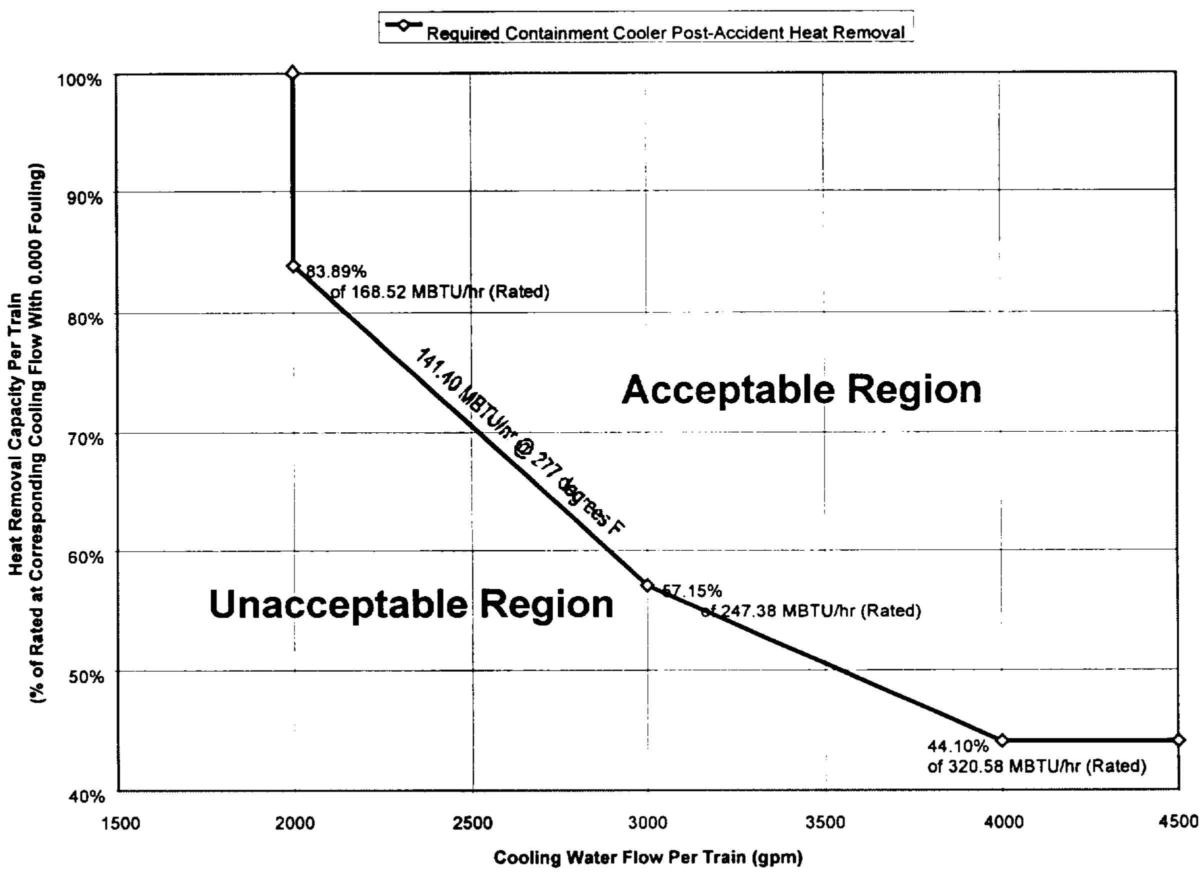

Figure B 3.6.6-1 (page 1 of 1)
Containment Cooler Heat Removal
Minimum Cooling Flow Rate

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Recirculation Fluid pH Control System

BASES

BACKGROUND

The Recirculation Fluid pH Control System (RFPC) is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the recirculated spray and to maximize retention of volatile iodine species in the sumps, the sump solution is adjusted to a minimum equilibrium sump pH of 7.1 A pH of greater than 7.0 minimizes the evolution of volatile iodine species from the sump solution as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The RFPC System includes stainless steel baskets containing trisodium phosphate crystalline (TSP-C). One such basket will be located within the confines of each containment recirculation sump. The baskets contain sufficient TSP-C to ensure a minimum equilibrium sump pH of 7.1. One seismically designed TSP-C basket is within the confines of each of the two containment recirculation sumps. Each basket is designed to contain a maximum of 6720 lbm of TSP-C (basis for the maximum depth of 36.8" in the Technical Specification) whereas a minimum depth of 30", corresponding to 4500 lbm, must be contained in each basket to ensure an equilibrium sump pH of at least 7.1. The baskets are located at an elevation that will ensure dissolution by the sump fluids. The baskets have a stainless steel frame with walls constructed of stainless steel grating and lined with #100 wire mesh stainless steel screening. Inside nominal dimensions of each basket is 80" x 56" x 38". The calculation of the minimum and maximum depths of TSP-C includes conservative allowances for compaction, spillage through the wire mesh, density variations, and the limited transformation of TSP-C into disodium triphosphate which is a weaker base (expected to have a small impact in the outer surface layer). The minimum equilibrium sump pH of 7.1 corresponds to a minimum of 9000 lbm of TSP-C in the baskets and a maximum sump boron concentration of 2500 ppm. If the maximum of 13,440 lbm of TSP-C were contained in the baskets at the end of cycle life such that a minimum sump boron concentration of 2007 ppm would occur, the maximum equilibrium sump pH would be less than 8.1.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The RFPC System is essential to the removal and retention of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that more than 90% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the RFPC System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System is inoperable.

The RFPC System satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RFPC System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume of TSP-C must be sufficient to raise the average long term containment sump solution pH to a level conducive to iodine removal and retention, namely, to greater than 7.1. This pH level maximizes the effectiveness of the iodine removal and retention mechanisms without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the RFPC System. The RFPC System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the RFPC System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the RFPC System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal/retention enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment

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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3.2 (Ref. 1). The MSSVs must have sufficient flow capacity to limit the secondary system pressure to $\leq$ 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 all valves following a turbine 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 $\leq$ 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, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump typically is the limiting AOO with respect to secondary system pressure. 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 power operated relief valves and sprays. 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES

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. 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 more than one MSSV on a single steam generator is 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 multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

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 (c)(2)(ii).

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to limit secondary system pressure. In MODES 4, 5 and 6 there are no credible transients requiring the MSSVs.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

In the case of only a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is not positive a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours. Required Action A.1 is only applicable when the Moderator Temperature Coefficient is negative at all power levels.

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.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Continued operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

B.1 and B.2

In the case of multiple inoperable MSSVS on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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. Furthermore, for a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive at any power level the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour Completion Time for required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.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 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 Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.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," provides sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If THERMAL POWER or the Power Range Neutron Flux - High trip setpoints is not reduced within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

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 (Ref. 5).

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, the valves are reset to $\pm$ 1% during the Surveillance to allow for drift. The lift settings pressure shall 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, if necessary, to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.3.2, Main Steam Supply System - System Description.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section 15.2, Decrease in Heat Removal by the Secondary System.
4. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. Westinghouse Letter SCP-99-129, dated July 7, 1999.

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 auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Condenser Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSIV is a 28 inch gate valve with dual-redundant hydraulic actuators. The assumed single failure of one of the redundant actuators will not prevent the MSIV from closing.

The MSIVs close on a main steam isolation signal generated by low steam line pressure, high steam line negative pressure rate or High-2 containment pressure. The MSIVs fail as is on loss of control or actuation power.

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, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6.2.1.4 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.1.5 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at 25% with loss of offsite power and the failure of one emergency diesel generator (Ref. 6). Because of increased energy storage in the primary plant, increased heat transfer in the steam generators, and the additional energy generation in the nuclear

(continued)

BASES

BACKGROUND (continued)

ESFAS channels. A single active failure in one power train would not prevent the other power train from functioning. The MFIVs provide the primary success path for events requiring feedwater isolation and isolation of non-safety-related portions from the safety-related portion of the system, such as, for auxiliary feedwater addition.

A description of the MFIVs and MFRVs is found in the FSAR, Section 10.4.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Credit is taken in accident analysis for the MFIVs to close on demand. The function of the MFRVs and associated bypass valves as discussed in the accident analysis is to provide a diverse backup function to the MFIVs for the potential failure of an MFIV to close even though the MFRVs are located in the non-safety-related portion of the feedwater system. Further assurance of feedwater flow termination is provided by the SGFP trip function; however, this is not credited in accident analysis. The accident analysis credits the main feedwater check valves as backup to the MFIVs to prevent SG blowdown for pipe ruptures in the non-seismic Category I portions of the feedwater system outside containment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii) indicates that components that are part of the primary success path and that actuate to mitigate an event that presents a challenge to a fission product barrier should be in Technical Specifications. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria) so that the plant response to the event remains within appropriate acceptance criteria. The primary success path does not include backup and diverse equipment. The MFIVs, with their dual-redundant actuators, are the primary success path for feedwater isolation; the MFRVs, bypass valves, and the SGFP trip function are backup and diverse equipment. Therefore, only the MFIVs are incorporated into Technical Specifications. The MFIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators, following an FWLB or 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 be OPERABLE. The MFIVs are considered OPERABLE when isolation times are within limits when given a fast close signal and they are capable of closing on an isolation actuation signal.

(continued)

BASES

LCO (continued)	Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event and failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MFIVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the MFIVs are closed they are performing their safety function. In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs are not required to mitigate the effects of a feedwater or steamline break in these MODES.
ACTIONS	The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve. <u>A.1 and A.2</u> With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 4 hours. When these valves are closed, they are performing their required safety function. The 4 hour Completion Time takes into account the redundancy afforded by the dual-redundant actuators on the MFIVs and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 4 hour Completion Time is reasonable, based on operating experience. Inoperable MFIVs that are closed must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MFIV(s) cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 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 3.7.3.1

This SR verifies that the closure time of each MFIV is $\leq$ 5 seconds from each actuator train when tested pursuant to the Inservice Testing Program. The MFIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. 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.

The Frequency for this SR is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at nominal operating temperature and pressure, as discussed in Reference 2. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.3.2

This SR verifies that each MFIV is capable of closure on an actual or simulated actuation signal. The manual fast close handswitch in the Control Room provides an acceptable actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage in conjunction with SR 3.7.3.1. However, it is acceptable to perform this surveillance individually.

The frequency of MFIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. This Frequency is acceptable from a reliability standpoint. This SR is modified by a NOTE that allows entry into and operation in MODE 3 prior to performing the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.3.2 (continued) This allows a delay of testing until MODE 3, to establish conditions consistent with those necessary to perform SR 3.7.3.1 and SR 3.7.3.2 concurrently.
REFERENCES	1. FSAR, Section 10.4.7, Condensate and Feedwater System. 2. ASME, Boiler and Pressure Vessel Code, Section XI. 3. FSAR, Table 7.3-14, NSSS Instrument Operating Conditions for Isolation Functions.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Steam Dump Valves (ASDs)

BASES

BACKGROUND

The ASDs, provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump Valves to the condenser not be available, as discussed in the FSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ASDs assure that subcooling can be achieved to facilitate equalizing pressure between the reactor coolant system and the ruptured steam generator following a postulated steam generator tube rupture event. The ASDs 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 Steam Dump System.

One ASD line for each of the four steam generators is provided. Each ASD line consists of one ASD and an associated manual isolation valve.

The ASDs are provided with upstream manual isolation valves to provide positive shutoff capability should an ASD develop seat leakage and to facilitate maintenance activities. The ASDs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ASDs are provided with a pressurized gas supply of nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ASDs. One nitrogen accumulator supplies one ASD and one auxiliary feedwater control valve per steam generator. The nitrogen accumulator supply is sized to provide sufficient pressurized gas to operate the ASD for the time required for Reactor Coolant System cooldown to RHR entry conditions.

A description of the ASDs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the ASDs is established by the capability to cool the unit to RHR entry conditions. The unit can be cooled to RHR entry conditions with only one steam generator and one ASD, utilizing the cooling water supply available in the CST. The valves will pass sufficient flow at all pressures to achieve a 50°F per hour plant cooldown rate. The total capacity of the four valves is 15% of rated main steam flow at steam generator no-load pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Reference 2, the ASDs 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. 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 in Reference 3, the operator is also required to perform a rapid cooldown using two intact steam generators 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 ASDs. The number of ASDs required to be OPERABLE to satisfy the SGTR accident analysis requirements is four. If a single failure of one occurs and another is associated with the ruptured SG, two ASDs would remain OPERABLE for heat removal.

The ASDs are equipped with manual isolation valves in the event an ASD spuriously fails to open or fails close during use.

The ASDs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Four ASD lines are required to be OPERABLE. One ASD line is required from each of four steam generators to ensure that at least two intact SG ASD lines are available to conduct the rapid RCS cooldown following an SGTR, in which one steam generator becomes unavailable due to the steam generator tube rupture, accompanied by a single, active failure of a second ASD line on an unaffected steam generator. The manual isolation valves must be OPERABLE to isolate a failed open ASD line. The accident analyses that credit OPERABILITY of the ASDs require them to relieve steam to the atmosphere in order to perform their safety related function.

Failure to meet the LCO can result in the inability to achieve subcooling, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the reactor coolant system and the ruptured steam generator.

An ASD is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and is capable of fully opening and closing on demand and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criterion, exists when conditions dictate closure of the manual isolation valve to limit leakage.

(continued)

BASES

LCO (continued)

Each nitrogen accumulator tank supplies one TDAFP control valve and one steam generator atmospheric steam dump valve. The tanks must be maintained at a pressure sufficient to ensure a five hour supply for the ASDs and the TDAFP flow control valves to be considered OPERABLE. The five hour supply is the minimum required for mitigation of a SBO or SGTR (Ref. 4).

APPLICABILITY

In MODES 1, 2, and 3, the ASD lines are required to be OPERABLE.

In MODE 4, the pressure and temperature limitations are such that the probability of a SGTR event requiring ASD operation is low. In addition, the RHR system is available to provide the decay heat removal function in MODE 4. Therefore, the ASD lines are not required to be OPERABLE in MODE 4.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ASD line inoperable for reasons other than excessive ASD seat leakage, action must be taken to restore the ASD line to OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ASD lines, a nonsafety grade backup in the Condenser Steam Dump System, and MSSVs. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two required ASD lines inoperable for reasons other than excessive ASD seat leakage, action must be taken to restore all but one required ASD line to OPERABLE status. Since the manual isolation valve can be closed to isolate an ASD, some repairs may be possible with the unit at power. The 72-hour Completion Time is reasonable to repair inoperable ASD lines, based on the availability of the Condenser Steam Dump System and/or MSSVs, and the low probability of an event occurring during the restoration period that would require the ASD lines.

C.1

With three or more required ASD lines inoperable for reasons other than excessive ASD seat leakage, action must be taken to restore all but two

(continued)

BASES

ACTIONS

C.1 (continued)

required ASD lines to OPERABLE status. Since the manual isolation valve can be closed to isolate an ASD, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ASD lines, based on the availability of the Condenser Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ASD lines.

D.1 and D.2

Requiring a 30 day limit for restoring an ASD valve to OPERABLE status from inoperable, due to excessive seat leakage from the valve, provides assurance that the required number of ASDs will be available for plant cooldown. This action limits the period in which a manual isolation valve is closed due to excessive seat leakage of the ASD and minimizes the delay associated with manually opening a closed manual isolation valve (due to excessive seat leakage of the ASD). Required ACTIONS D.1 and D.2 are modified by a Note indicating that LCO 3.0.4 does not apply

E.1 and E.2

If the required ASD line(s) 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 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 3.7.4.1

To perform a controlled cooldown of the RCS, the ASDs must be able to be opened remotely and throttled through their full range. This SR ensures that the ASDs are tested through a full control cycle as described in the Inservice Test Program.

The conditions that best verify the operability of the ASDs is with the manual isolation valve open and nominal steam line operating pressure and temperature. The ASDs are designed such that steam line pressure acts on top of the valve plug. When the valve is required to move to the open position the actuator must act against steam line pressure. For this surveillance requirement to best verify the operational readiness of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 (continued)

ASDs, it should be performed at nominal SG operating temperature and pressure, which is in the upper portion of MODE 3 (Ref. 5).

Use of an ASD during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the required Inservice Testing Program Frequency. The Frequency is acceptable from a reliability standpoint.

This Surveillance Requirement is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which acceptance criterion was generated (Ref. 5).

SR 3.7.4.2

The function of the manual isolation valve is to isolate a failed open or leaking ASD. Cycling the manual isolation valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the manual isolation valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed in accordance with the Inservice Testing Program Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.3, Main Steam Supply System.
2. FSAR, Chapter 15.2, Decrease in Heat Removal by the Secondary System.
3. FSAR, Section 15.6.3, Steam Generator Tube Failure.
4. FSAR, Section 9.3.1, Compressed Air System.
5. Operating License Amendments 45 and 59.
6. Operating License Amendment 131.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps normally take suction through a common suction line from the condensate storage tank (CST) (LCO 3.7.6). Should the CST become unavailable, cooling water is available from the Essential Service Water (ESW) system. Each motor-driven AFW pump is supplied from one ESW train. The steam turbine driven AFW pump is supplied from either ESW train. The AFW pumps discharge 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 atmospheric steam dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the Condenser Steam Dump valves and condensate recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of the feedwater flow required for removal of decay heat from the reactor assumed in the accident analyses. The turbine driven pump provides 200% of the capacity of a motor driven pump. The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be locally aligned to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves and water from either the condensate storage tank or redundant ESW supply lines. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. In addition, each of the ESW supply lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with normally open air operated control

(continued)

BASES

LCO (continued)

redundant ESW supply lines and supplying AFW to any of the steam generators. The inoperability of a single supply line from an ESW train to the TDAFP does not render the TDAFP inoperable. Therefore, with one ESW train inoperable, the associated MDAFP train is considered inoperable and one TDAFP supply line is considered inoperable. However, the TDAFP is OPERABLE based on the remaining OPERABLE ESW supply line. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. At least one condensate drain path, via FCLV0010 or FCST0001, must exist for the TDAFP to be OPERABLE.

The standby lineup for the TDAFP steam supply lines is when the main steam supply valves, ABHV0005 and ABHV0006, are closed and OPERABLE and the warmup valves, ABHV0048 and ABHV0049, are open and OPERABLE. The TDAFP steam supply lines may also be considered OPERABLE when the associated main steam supply and warmup valves are failed or secured in their safeguards position.

Each nitrogen accumulator tank supplies one TDAFP control valve and one steam generator atmospheric steam dump valve. The tanks must be maintained at a pressure sufficient to ensure a five hour supply for the TDAFP flow control valves and the ASDs to be OPERABLE. The five hour supply is the minimum required for mitigation of a Station Blackout (SBO) or SGTR (Ref. 3).

Although the AFW system can be used in MODE 4 to remove decay heat, the LCO does not require the AFW system to be OPERABLE in MODE 4 since the RHR system is available for decay heat removal.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODES 4 and 5, the AFW System may be used for heat removal via the steam generators but is not required since the RHR System is available in this MODE.

In MODE 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the two Essential Service Water supply lines in the turbine driven AFW train inoperable, action must be taken to restore the inoperable ESW supply line to OPERABLE status within 72 hours. One inoperable ESW supply line in the turbine driven AFW train does not render TDAFP inoperable since the turbine driven AFW train is provided with redundant ESW supply lines. The 72 hour Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE Essential Service Water supply line in the turbine driven AFW train;

b. The availability of the preferred nonsafety grade Condensate Storage Tank supply;

(continued)

BASES

ACTIONS

B.1 (continued)

c. The availability of at least one OPERABLE motor driven AFW pump. When an ESW train inoperability renders a TDAFP supply line inoperable and a motor driven AFW pump supply line inoperable, then one motor driven AFW pump is OPERABLE and the second motor driven AFW pump is available with water supplied from the nonsafety grade Condensate Storage Tank;

d. The low probability of an event occurring that will require the inoperable Essential Service Water supply line to the turbine driven AFW pump; and

e. The 72 hour Completion Time is consistent with the allowed Completion Time for one train of ESW inoperable.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1

With one of the required AFW trains (pump or flow path) inoperable for reasons other than Condition A or Condition B, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines or two ESW supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered concurrently. The AND connector between 72 hours and 10 days

(continued)

BASES

ACTIONS

C.1 (continued)

dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

D.1 and D.2

When Required Action A.1 or B.1 or C.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 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.

E.1

If all three AFW trains are inoperable, 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 AFW train to OPERABLE status.

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. 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 AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves capable of being

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves and relief valves. Additionally, vent and drain valves are not within the scope of this SR.

This SR is modified by a Note indicating that the SR is not required to be performed for the AFW flow control valves until the AFW system is placed in automatic control or when Thermal Power is above 10% RTP.

In order for the TDAFP and MDAFPs to be OPERABLE while the AFW system is in automatic control or above 10% RTP, the discharge flow control valves (ALHV0005, 6, 7, 8, 9, 10, 11, and 12) shall be in the full open position. The TDAFP and MDAFPs remain OPERABLE with the discharge flow control valves throttled to maintain steam generator levels during plant heatup, cooldown, or if started due to an Auxiliary Feedwater Actuation Signal (AFAS) or manually started in anticipation of an AFAS.

The 31 day Frequency, based on engineering judgment, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. Testing of each MDAFP on a recirculation flow rate greater than or equal to 75 gpm and ensuring a discharge pressure of greater than or equal to 1535 psig verifies the capability of each MDAFP to deliver a total pump flow of 575 gpm at a steam generator pressure of 1221 psig. The capability to deliver 575 gpm includes the 75 gpm recirculation flow plus 250 gpm delivered to two steam generators from one MDAFP. Testing the TDAFP on a recirculation flow greater than or equal to 120 gpm and ensuring a discharge pressure of greater than or equal to 1625 psig verifies the capability of the pump to deliver a total pump flow of 1145 gpm at a steam generator pressure of 1221 psig. The capability to deliver 1145 gpm includes the 120 gpm recirculation flow, 25 gpm to pump auxiliary loads, and 250 gpm delivered to four steam generators from the TDAFP. Such

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) satisfies this requirement. The test frequency in accordance with the Inservice Testing Program results in testing each pump once every 3 months, as required by Reference 2.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment

This SR includes the requirement to verify that each AFW motor-operated discharge valve, ALHV0005, 7, 9 and 11, limits the flow from the motor-driven pump to each steam generator to $\leq$ 320 gpm and that valves ALHV0030, 31, 32, 33, 34, 35 and 36 actuate to the required position upon receipt of an Auxiliary Feedwater Pump suction Pressure-Low signal.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an AFAS by demonstrating that each AFW pump starts automatically on an actual or simulated auxiliary feedwater actuation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

This SR is modified by a Note. The Note indicates that the SRbe deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6.

OPERABILITY of AFW flow paths must beverified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System 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 to the steam generators is properly aligned.

REFERENCES

1. FSAR, Section 10.4.9, Auxiliary Feedwater System.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. FSAR, Section 9.3.1, Compressed Air System.
4. Amendment No. 55 to facility Operating License No. NPF-30, dated 7/27/90.
5. FSAR 15.2.8, Feedwater System Pipe Break.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a nonsafety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric steam dump valves. The AFW pumps operate with a continuous recirculation to the CST.

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 condenser steam dump valves. The condensed steam can be returned to the CST by the condensate pumps. This has the advantage of conserving condensate while minimizing releases to the environment.

The CST capacity allows the plant to remove decay heat from the primary system during a 4 hour Station Blackout event (Ref. 3). However, the CST is not the safety-related source of water to the AFW pumps. The safety-related source is provided by the essential service water (ESW) system (LCO 3.7.8).

A description of the CST is found in the FSAR, Section 9.2.6 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST is the preferred suction supply to the Auxiliary Feedwater Pumps (AFP) due to the quality of the water. However, the CST is a nonseismic structure and thus cannot be relied upon for all accident scenarios. In order to ensure a safety grade supply of water is available to supply the suction of the AFP's for all credible accident conditions, the Essential Service Water System has been designed to provide the backup emergency supply to the AFP's.

Since the CST is the preferred source for the AFP's, it has been designed with sufficient capacity to cooldown the RCS from Hot Standby to Hot Shutdown during a 4 hour Station Blackout Event (Ref. 3). The contained water volume limit includes an allowance for water not useable because of tank discharge line location or other physical characteristics. Additional details regarding the design of the AFW system can be found in FSAR 10.4.9.

(continued)

BASES

ACTIONS

A.1 (continued)

be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW 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 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path servicing safety related equipment provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control. This SR does not apply to valves that cannot be inadvertently misaligned,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

such as check valves and relief valves. Additionally, vent and drain valves are not within the scope of this SR.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves, servicing safety related components or isolating nonsafety related components, on an actual or simulated actuation signal. This SR applies to the CCW valves that receive a Safety Injection signal and the RCP thermal barrier valves that receive a high CCW flow 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 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. These actuation signals include Safety Injection and Loss of Power. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves and relief valves. Additionally, vent and drain valves are not within the scope of this SR.

The 31 day Frequency is based on engineering judgment, and is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the ESW system valves servicing safety related components or isolating the nonsafety related components on an actual or simulated actuation signal. These actuation signals include Loss of Power, SIS, and Low AFW Suction Pressure. The ESW system is a standby emergency system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the ESW system pumps on an actual or simulated actuation signal. These actuation signals include SIS, Low AFW Suction Pressure, and Loss of Power. The ESW system is a standby emergency system that cannot be fully actuated as part of normal testing during normal operation. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. The ESW pump start on low AFW Suction Pressure Surveillance is performed under the conditions that apply during a unit outage and has the potential for an unplanned transient if the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.3 (continued)

Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.1, Essential Service Water System.
2. FSAR, Section 6.2, Containment Systems.
3. FSAR, Section 5.4.7, Residual Heat Removal System.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CREVS consists of two independent, redundant trains that pressurize, recirculate, and filter the control room air. Each CREVS train consists of a filtration system train and a pressurization system train. Each filtration system train consists of a fan, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a second HEPA filter follows the adsorber section to collect carbon fines. Each pressurization system train consists of a fan, a moisture separator, an electric heater, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a second HEPA filter follows the adsorber section to collect carbon fines. Ductwork, valves or dampers, and instrumentation also form part of the CREVS system.

The CREVS is an emergency system which may also operate during normal unit operations. Upon receipt of the actuating signal, normal air supply and exhaust to the control room is isolated, and a portion of the ventilation air is recirculated through the system filter trains, and the pressurization system is started. The prefilters remove any large particles in the air, and a moisture separator removes any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each pressurization system train for at least 10 hours per month, with the heaters functioning, reduces moisture buildup on the HEPA filters and adsorbers. The heaters are important to the effectiveness of the charcoal adsorbers.

Actuation of the CREVS by a Control Room Ventilation Isolation Signal (CRVIS), places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The emergency (CRVIS) mode also initiates pressurization and filtered ventilation of the air supply to the control room.

The control room pressurization system draws in outside air, processing it through a particulate filter charcoal adsorber train for cleanup. This outside air is diluted with air drawn from the cable spreading rooms and the electrical equipment floor levels and distributed back into those spaces for further dilution. The control room filtration units take a portion

(continued)

BASES

BACKGROUND (continued)

of air from the exhaust side of the pressurization system, upstream of the outside air intake, for dilution with portions of the exhaust air from the control room air-conditioning system and processes it through the control room filtration system adsorption train for additional cleanup. This air is then further diluted with the remaining control room air-conditioning system return air, cooled, and supplied to the control room. This process will maintain the control room under a positive pressure of 1/8 inch water gauge (min.) with respect to the outside atmosphere. This will assure exfiltration from the control room, thus preventing any unprocessed contaminants from entering the control room.

The air entering the control building during normal operation is continuously monitored by radiation, carbon dioxide/monoxide, and smoke detectors. A high radiation signal initiates the emergency (CRVIS) mode of operation; the other detectors provide an alarm in the control room. A CRVIS is initiated by the radiation monitors (GKRE0004 and GKRE0005), Fuel Building Ventilation Isolation Signal, Containment Isolation Phase A, the containment purge exhaust radiation monitors (GTRE0022 and GTRE0033), and manually. The instrumentation associated with actuation of the CREVS is addressed in LCO 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation."

A single train is capable of pressurizing the control room to $\geq$ 0.125 inches water gauge. The CREVS operation in maintaining the control room habitable is discussed in the FSAR, Section 6.4 (Ref. 1).

Redundant pressurization and filtration 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 CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

coolant accident, fission product release presented in the FSAR, Chapter 15A.3 (Ref. 2).

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREVS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release.

The CREVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CREVS train is OPERABLE when the associated:

a. Control Room Air Conditioner, filtration and pressurization fans are OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

Note that the Control Room Air Conditioning System (CRACS) forms a subsystem to the CREVS. The CREVS remains capable of performing its safety function provided the CRACS air flow path is intact and air circulation can be maintained. Isolation or breach of the CRACS air flow

(continued)

BASES (continued)

APPLICABILITY

path can also render the CREVS flow path inoperable. In these situations, LCOs 3.7.10 and 3.7.11 may be applicable.

In MODES 1, 2, 3, 4, 5, and6, and during movement of irradiated fuel assemblies, CREVS must be OPERABLE to control operator exposure during and following a DBA.

In MODE 5 or 6, the CREVS is required to cope with the design basis release from the rupture of a waste gas tank.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a design basis fuel handling accident.

ACTIONS

A.1

When one CREVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREVS train could result in loss of CREVS 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

If the control room boundary is inoperable in Modes 1, 2, 3, and 4 such that neither CREVS train can establish the required pressure, action must be taken to restore an OPERABLE control room boundary within 24 hours. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the availability of the CREVS to provide a filtered environment (albiet with potential control room inleakage).

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or control room 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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1.1, D.1.2, D.2.1, and D.2.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the CRVIS mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

Action D.1.2 requires the CREVS train placed in operation be capable of being powered by an emergency power source. This action assures OPERABILITY of the CREVS train in the unlikely event of a Fuel Handling Accident or Decay Tank rupture while shutdown concurrent with a loss of offsite power.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1 and E.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREVS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or4, for reasons other than an inoperable control room boundary (i.e., Condition B), the CREVS 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.

(continued)

BASES (continued)

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 severe, testing each train once every month, by initiating from the control room, flow through the HEPA filters and charcoal adsorbers of both the filtration and pressurization systems, provides an adequate check of this system.

Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each pressurization system train must be operated for $\geq$ 10 continuous hours with the heaters functioning. Functioning heaters will not necessarily have the heating elements energized continuously for 10 hours; but will cycle depending on the air temperature. Each filtration system train need only be operated for $\geq$ 15 minutes to demonstrate the function of the system. The31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

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

The CREVS filter tests use the test procedure guidance in Regulatory Guide 1.52 (Ref. 3). 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 CREVS train starts and operates on an actual or simulated actuation signal. The actuation signal includes Control Room Ventilation Isolation or High Gaseous Radioactivity. The CREVS train automatically switches on an actual or simulated CRVIS signal into a CRVIS mode of operation with flow through the HEPA filters and charcoal adsorber banks. The Surveillance Requirement also verifies that a control room ventilation isolation signal (CRVIS) will be received by the LOCA sequencer to enable an automatic start of the Diesel Generator loads that are associated with a CRVIS. Verification that these loadswill start and operate at the appropriate step in the LOCA sequencer and that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.3 (continued)

other auto-start signals for these loads will be inhibited until the LOCA sequencer is reset is accomplished under Surveillance Requirement SR 3.8.1.12. The Frequency of 18 months is consistent with the typical operating cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to the outside atmosphere, is periodically tested to verify proper functioning of the CREVS. During the CRVIS mode of operation, the CREVS is designed to pressurize the control room $\geq$ 0.125 inches water gauge positive pressure with respect to the outside atmosphere in order to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. FSAR, Section 6.4, Habitability Systems.

2. FSAR, Chapter 15A.3, Control Room Radiological Consequences Calculation Models.

3. Regulatory Guide 1.52, Rev. 2, Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants.

4. NUREG-0800, Section 6.4, Rev. 2, July 1981, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning System (CRACS)

BASES

BACKGROUND	The CRACS provides temperature control for the control room. The CRACS consists of two independent and redundant trains that provide cooling of recirculated control room air. Each train consists of a prefilter, self-contained refrigeration system (using essential service water as a heat sink), centrifugal fans, instrumentation, and controls to provide for control room temperature control. The CRACS is a subsystem to the CREVS, described in LCO 3.7.10, providing air temperature control for the control room. The CRACS is an emergency system, which also operates during normal unit operations. A single train will provide the required temperature control to maintain the control room ≤ 84°F. The CRACS operation in maintaining the control room temperature is discussed in the FSAR, Section 9.4.1 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the CRACS is to maintain the control room temperature for 30 days of continuous occupancy. The CRACS components are arranged in redundant, safety related trains. During normal or emergency operations, the CRACS maintains the temperature ≤ 84°F. A single active failure of a component of the CRACS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY. The CRACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

(continued)

BASES

LCO (continued)

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the refrigeration compressors, heat exchangers, cooling coils, fans, and associated temperature control instrumentation. In addition, the CRACS must be operable to the extent that air circulation can be maintained. Isolation or breach of the CRACS air flow path can also render the CREVS flowpath inoperable. In these situations, LCO 3.7.10 would also be applicable.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements.

ACTIONS

A.1

With one CRACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in a loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation and the consideration that the remaining train can provide the required protection.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1.1, C.1.2, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur,

(continued)

BASES

ACTIONS

C.1.1, C.1.2, C.2.1, and C.2.2 (continued)

and that active failures will be readily detected. Action C.1.2 requires the CRACS train placed in operation be capable of being powered by an emergency power source. This action assures OPERABILITY of the CRACS train in the unlikely event of a Fuel Handling Accident or Decay Tank rupture while shut down concurrent with a loss of offsite power.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRACS trains are inoperable in MODE 1, 2, 3, or 4, the CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the CRACS air conditioning units is adequate to remove the heat load assumed in the control room during design basis accidents. This SR consists of verifying the heat removal capability of the condenser heat exchanger (either through performance testing or inspection), ensuring the proper operation of major components in the refrigeration cycle and verification of unitair flow capacity. The 18 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section 9.4.1, Control Building HVAC.

BASES

APPLICABLE SAFETY ANALYSES (continued)

all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) and Containment Spray System during the recirculation mode are filtered and adsorbed by the Emergency Exhaust System. The DBA analysis of the fuel handling accident and of the LOCA assumes that only one train of the Emergency Exhaust System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel building is determined for a fuel handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guides 1.4 (Ref. 6) and 1.25 (Ref. 5).

The Emergency Exhaust System satisfies Criterion 3 of 10CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant trains of the Emergency Exhaust System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the auxiliary building or fuel building exceeding regulatory release limits in the event of a LOCA or fuel handling accident.

In MODES 1, 2, 3 and 4 the Emergency Exhaust System (EES) is considered OPERABLE when the individual components necessary to control releases from the auxiliary building are OPERABLE in both trains (i.e., the components required for the SIS mode of operation and the auxiliary building pressure boundary). During movement of irradiated fuel assemblies in the fuel building, the EES is considered OPERABLE when the individual components necessary to control releases from the fuel building are OPERABLE in both trains (i.e. the components required for the FBVIS mode of operation and the fuel building pressure boundary). An Emergency Exhaust System train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

(continued)

BASES

LCO (continued)	The LCO is modified by a Note allowing the auxiliary or fuel building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary or fuel building isolation is indicated.
APPLICABILITY	In MODE 1, 2, 3, or 4, the Emergency Exhaust System is required to be OPERABLE to support the SIS mode of operation to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus. In MODE 5 or 6, the Emergency Exhaust System is not required to be OPERABLE since the ECCS is not required to be OPERABLE. During movement of irradiated fuel in the fuel building, the Emergency Exhaust System is required to be OPERABLE to support the FBVIS mode of operation to alleviate the consequences of a fuel handling accident. The Applicability is modified by a Note. The Note clarifies the Applicability for the two safety-related modes of operation of the Emergency Exhaust System, i.e., the Safety Injection Signal (SIS) mode and the Fuel Building Ventilation Isolation Signal (FBVIS) mode. The SIS mode which aligns the system to the auxiliary building is applicable when the ECCS is required to be OPERABLE. In the FBVIS mode the system is aligned to the fuel building. This mode is applicable while handling irradiated fuel in the fuel building.
ACTIONS	A.1 With one Emergency Exhaust System train inoperable in MODE 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the Emergency Exhaust System function. This condition only applies to the EES components required to support the SIS mode of operation. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable Emergency Exhaust System train, and the remaining Emergency Exhaust System train providing the required protection.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.14 Not used.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel in the fuel building, the water level in the fuel storage pool is an initial condition design parameter in the analysis of the fuel handling accident as postulated by Reg. Guide 1.25 (Ref. 4). Irradiated fuel being moved is assumed to be from a reactor core which has been subcritical for at least 100 hours. A minimum water level of 23 feet (Regulatory Position C.1.c of Ref. 4) allows a decontamination factor of 100 (Regulatory Position C.1.g) of Ref.4 to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of the damaged rods is retained by the fuel storage pool water. The fission product release point is assumed to be at the point of impact at the top of the spent fuel storage racks. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref.4).

The fuel handling accident inside the fuel building is described in Reference 3. With a minimum water level of 23 feet and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within the limits of 10CFR 100 (Refs. 4 and 5).

The fuel storage pool water level satisfies Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The fuel storage pool water level is required to be ≥ 23 ft over the top of the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists. The reconstitution of irradiated fuel assemblies is also considered movement of irradiated fuel.

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 pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. 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 3.7.15.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling pool, and the level in the refueling pool is checked daily in accordance with SR 3.9.7.1.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.2, Spent Fuel Storage.
2. FSAR, Section 9.1.3, Fuel Pool Cooling and Cleanup System.
3. FSAR, Section 15.7.4, Fuel Handling Accidents.
4. Regulatory Guide 1.25, Rev. 0, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility For Boiling and Pressurized Water Reactors.
5. 10 CFR 100.11.
6. NUREG-0800, Section 15.7.4, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.16 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

In the High Density Rack (HDR) design (Refs. 1 and 2), each fuel pool storage rack location is designated as either Region 1, Region 2, Region 3, or empty (in the checkerboarding configuration). Numerous configurations of region designation are possible. Criteria are established for determining an acceptable configuration (Ref. 1). The HDRs will store a maximum of 2363 fuel assemblies in the spent fuel pool and potentially an additional 279 fuel assemblies in the cask loading pool (with racks installed). Full-core offload capability will be maintained. The fuel storage pool consists of the spent fuel pool and the cask loading pool (with racks installed). Region 1 locations are designed to accommodate new fuel with a nominal maximum enrichment of 4.6 wt% U-235 with no integral fuel burnable absorber (IFBA); or up to a nominal maximum enrichment of 5.0 wt% U-235 with 16 IFBA; or spent fuel which meets the requirements of paragraph 4.3.1.1 in Section 4.3. Region 2 and 3 locations are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage. Locations designated as empty cells shall contain no fuel assemblies.

The water in the fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the HDR design is based on the use of unborated water, which maintains fuel storage pool in a subcritical condition during normal operation with the fuel storage pool racks fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of multiple fuel assemblies in non-region 1 locations. This could potentially increase the reactivity of the fuel storage pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the HDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly

(continued)

BASES

BACKGROUND (continued)	Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.
APPLICABLE SAFETY ANALYSES	Accidents can be postulated that could increase the reactivity of the fuel storage pool which are unacceptable with unborated water in the fuel storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool maintains subcriticality with K_{eff} of 0.95 or less. The postulated accidents are basically of two types. Multiple fuel assemblies could be incorrectly transferred to non-region 1 locations (e.g., unirradiated fuel assemblies or insufficiently depleted fuel assemblies). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded storage rack. The negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the FASR, Appendix 9.1A (Ref. 1). Safety analyses assume a B-10 enrichment of 19.9 a/o (Ref. 1). Administrative controls on the soluble boron concentration in the fuel storage pool ensure that there is equivalent B-10 concentration. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	The fuel storage pool boron concentration is required to be $\geq$ 2165 ppm. The fuel storage pool consists of the spent fuel pool and cask loading pool (with racks installed). The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 1. This concentration of dissolved boron is the minimum required concentration for non-inventoried fuel assembly storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the fuel storage pool, until a complete fuel storage pool verification has been performed following the last movement of fuel assemblies in the fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for misloaded fuel assemblies or a dropped fuel assembly.

(continued)

BASES (continued)

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 pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. 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 fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. Callaway FSAR, Appendix 9.1A, "The High Density Rack (HDR) Design Concept."

2. Amendment No. 129 dated January 19, 1999 to the Callaway Operating License.

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The high density rack modules for the fuel storage pool are designed for storage of both new fuel and spent fuel. Spent fuel storage is designated into Regions based upon initial enrichment and accumulated burnup. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.6 wt% U-235 with no burnable absorbers or up to 5.0 wt% U-235 with integral absorbers. Region 2 and Region 3 are designed to accommodate fuel of up to 5.0 wt% U-235 initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO.

Prior to storage of fuel assemblies in the fuel storage pool, overall pool storage configurations are prepared in accordance with administrative controls. The pool layouts include sufficient Region 1 storage to accommodate new and discharged fuel assemblies with low burnup. Fuel storage utilizes either a Mixed Zone Three Region configuration and/or a checkerboarding configuration.

In a Mixed Zone Three Region configuration, Region 1 storage cells are only located along the outside periphery of the rack modules and must be separated by one or more Region 2 storage cells. Region 1 storage cells may be located directly across from one another when separated by a water gap. The outer rows of alternating Region 1 and 2 storage cells must be further separated from the internal Region 3 storage cells by one or more Region 2 storage cells.

In the checkerboarding configuration, fuel assemblies are placed in an alternating checkerboard style pattern with empty storage cells (i.e., fuel assemblies are surrounded on all four sides by empty storage cells except at the checkerboard boundary). Region 1 fuel assemblies may not be located directly across from one another, even when separated by a water gap. This arrangement may be used anywhere in the fuel storage area and may be combined with the Mixed Zone Three Region configuration within a rack module, if the checkerboarding pattern is maintained in a linear array equal to or greater than 2 x 2. A checkerboard area may be bounded by either a water gap, empty cells, Region 2 fuel assemblies or Region 3 fuel assemblies.

The water in the fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost,

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.18 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.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

Operating a unit at the allowable secondary coolant specific activity will assure that the potential 2 hour exclusion area boundary (EAB) exposure is limited to a small fraction of the 10CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15.1.5 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration greater than 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for wholebody and thyroid dose rates.

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 steam generator atmospheric steam dump valves (ASDs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and (ASDs) during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, both the RCS and the steam generators are at reduced pressure or are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed

(continued)

BASES

BACKGROUND (continued)

Description	Approximate Crankcase Volume
Hi level alarm	1215 gallons
Dip stick "full" mark	1200 gallons
Automatic Makeup valve isolates	1116 gallons
Automatic Makeup valve opens	1063 gallons
Low level alarm	963 gallons
Dip stick "add oil" mark (10 day supply)	948 gallons
7 day supply	750 gallons
6 day supply	686 gallons
Unusable volume in crankcase	300 gallons

Each DG has an air start system with adequate capacity for five successive start attempts on the DG without recharging the air start receiver(s).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 4), and in the FSAR, Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section3.6, Containment Systems.

Since diesel 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(c)(2)(ii).

(continued)

BASES (continued)

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system is required to have a minimum capacity for five successive DG start attempts without recharging the air start receivers.

The diesel generators are not considered inoperable when the missile cover is removed from the emergency diesel fuel oil storage tanks so long as appropriate administrative controls are followed to ensure adequate missile protection is maintained. These controls include: limiting the time the cover may be removed to 36 consecutive hours, maintaining the cover rigged to the crane with personnel stationed to facilitate immediate replacement, and monitoring weather forecasts and local conditions to allow immediate replacement of the cover if necessary.

Both diesel generators are inoperable when the diesel fuel transfer systems are cross-connected, except in MODE 5 or 6 when cross-connected under administrative controls. Administrative controls consist of stationing operators at each cross-connect isolation valve (JEV0007 and JEV0008) with communications established between the operators and control room. In this way, the cross-connect isolation valves can be rapidly closed when a problem occurs with the cross connection. Cross-connecting the DGs in MODE 1, 2, 3 or 4 under administrative controls is not permitted, because the DGs would no longer be separate and independent as required by LCO 3.8.1.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and the starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the 7 day fuel oil supply for a DG 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 DG 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.

B.1

In this Condition there may not be an adequate amount of lube oil capacity to support the 7 days of full load operation of the diesel engine. However, in the unlikely event that this Condition is entered the lower limit was established to ensure that there was at least a 6 day supply of lube oil. 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 diesel generator inoperable.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the

(continued)

BASES

ACTIONS

C.1 (continued)

diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 435 psig in two receivers or < 610 psig in one receiver, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 250 psig in two receivers or ≥ 300 psig in one receiver, there is adequate capacity for at least one start attempt, and the DG 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 DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil, or starting air subsystem not within the SR limits and not within limits specified by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The750 gal requirement is based on the DG manufacturer consumption values for the run time of the DG. There are several methods available to verify a lube oil volume greater than or equal to 750 gallons. The preferred method is to verify that lube oil level is greater than the required inventory with the engine dipstick. Other indirect methods, such as the local level indicator or the absence of a low level alarm are acceptable as alternate methods.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057- (Ref. 6);

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F; and

c. Verify that the new fuel oil has a water and sediment content of less than or equal to 0.05% when tested in accordance with ASTM D1796-83 (Ref. 6).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 (Ref. 6), ASTM D2622-82 (Ref. 6) or ASTM D4294-90 (Ref. 6) and the acceptance criteria for fuel oil maximum cloud point is 0°C. If the31 day sample results from the vendor come back unsatisfactory per Table 1 of ASTM-D975-81, Condition D is entered. If a representative sample is taken from the underground fuel oil storage tank, is analyzed and passes the sample parameters in Table 1 of ASTM-D975-81 that had tested as unsatisfactory, Condition D is exited. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-78, Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. The filter size for the determination of particulate contamination will be 3.0 micron nominal instead of 0.8 micron nominal as specified by ASTM D2276-78, Method A (Ref. 6). It is acceptable to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

contamination will be 3.0 micron nominal instead of 0.8 micron nominal as specified by ASTM D2276-78, Method A (Ref. 6). It is acceptable to obtain a field sample for subsequent laboratory testing in lieu offield testing.

SR 3.8.3.4

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined as 3 seconds of cranking time or approximately 2 to 3 engine revolutions. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.6

This SR is not applicable.

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. ASTM Standards: D4057-; D975-81; D1796-83; D1552-79; D2622-82; D2276, Method A, D4294-90.
7. ASTM Standards, D975, Table 1.
8. ASME, Boiler and Presser Vessel Code, Section XI.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should 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.

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.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to trend overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer'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.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life the Surveillance Frequency is reduced to 18 months. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below 90% of the manufacturer's rating.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1978.
4. FSAR, Chapter 8.
5. IEEE-485-1983, June 1983.
6. FSAR, Chapter 6.
7. FSAR, Chapter 15.
8. Regulatory Guide 1.93, December 1974.
9. IEEE-450-1995.
10. Regulatory Guide 1.32, February 1977.
11. Regulatory Guide 1.129, February 1978.

BASES

APPLICABLE SAFETY ANALYSES (continued)	assure that the desired level of minimal risk is maintained (frequently referred to as maintaining a desired defense in depth). The level of detail involved in the assessment will vary with the significance of the equipment being supported. In some cases, prepared guidelines are used which include controls designed to manage risk and retain the desired defense in depth. The DC sources satisfy Criterion 3 of the 10 CFR 50.36(c)(2)(ii).
LCO	One DC electrical power subsystem shall be OPERABLE to support one train of the DC electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems - Shutdown." The required DC electrical power subsystem (Train A or Train B) shall consist of two DC buses energized from the associated batteries and chargers or swing charger powered from the respective Class 1E 480 V load center, and the corresponding control equipment and interconnecting cabling within the train. 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). The required DC electrical power distribution subsystem is supported by one train of DC electrical power system. When the second DC electrical power distribution train (subsystem) is needed to support redundant required systems, equipment and components, the second Train may be energized from any available source. The available source must be Class 1E or another reliable source. The available source must be capable of supplying sufficient DC electrical power such that the redundant components are capable of performing their specified safety function(s) (implicitly required by the definition of OPERABILITY). Otherwise, the supported components must be declared inoperable and the appropriate conditions of the LCOs for the redundant components must be entered.

(continued)

BASES

LCO
(continued)

TRAIN A		TRAIN B	
Bus NK01 energized from Battery NK11 and Charger NK21 or Swing Charger NK25 (powered from AC Load Center NG01)	Bus NK03 energized from Battery NK13 and Charger NK23 or Swing Charger NK25 (powered from AC Load Center NG01)	Bus NK02 energized from Battery NK12 and Charger NK22 or Swing Charger NK26 (powered from AC Load Center NG04)	Bus NK04 energized from Battery NK14 and Charger NK24 or Swing Charger NK26 (powered from AC Load Center NG04)

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 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.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

By allowing the option to declare required features inoperable with the associated DC power source inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In addition to the requirements established by the technical specifications, the plant staff must also manage shutdown tasks and electrical support to maintain risk at an acceptably low value.

As required by the technical specifications, one train of the required equipment during shutdown conditions is supported by one train of AC and DC power and distribution. The availability of additional equipment, both redundant equipment as required by the technical specifications and equipment not required by the specifications, contributes to risk reduction and this equipment should be supported by reliable electrical power systems. Typically the Class 1E power sources and distribution systems of the unit are used to power equipment because these power and distribution systems are available and reliable. When portions of the Class 1E power distribution systems are not available (usually as a result of maintenance or modifications), other reliable power sources or

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

distribution are used to provide the needed electrical support. The plant staff assesses these alternate power sources and distribution systems to assure that the desired level of minimal risk is maintained (frequently referred to as maintaining a desired defense in depth). The level of detail involved in the assessment will vary with the significance of the equipment being supported. In some cases, prepared guidelines are used which include controls designed to manage risk and retain the desired defense in depth.

The inverters satisfy Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

LCO

One train of inverters shall be OPERABLE to support one train of the onsite Class 1E AC vital bus electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems - Shutdown." The required train of inverters (Train A or Train B) shall consist of two AC vital buses energized from the associated inverters with each inverter connected to the respective DC bus. 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 4.16 kV 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 (e.g., fuel handling accidents).

The required AC vital bus electrical power distribution subsystem is supported by one train of inverters. When the second (subsystem) of AC vital bus electrical power distribution is needed to support redundant required systems, equipment and components, the second train may be energized from any available source. The available source must be Class 1E or another reliable source. The available source must be capable of supplying sufficient AC electrical power such that the redundant components are capable of performing their specified safety function(s) (implicitly required by the definition of OPERABILITY). Otherwise, the supported components must be declared inoperable and the appropriate conditions of the LCOs for the redundant components must be entered.

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B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of filled portions of the Reactor Coolant System (RCS) and the refueling pool that have direct access to the reactor vessel during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration is sufficient to maintain Shutdown Margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in MODE 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control cluster assembly completely removed from its fuel assembly.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the main system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling cavity is then flooded to form the refueling pool. Typically, the refueling pool is flooded with borated water from the refueling water storage tank through the open reactor vessel by the use of the Residual Heat Removal (RHR) System pumps or gravity feeding.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling pool mix the added concentrated boric acid with the water in the refueling pool. The RHR System is in operation during refueling (see 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") to provide forced circulation in the RCS

(continued)

BASES

BACKGROUND (continued)

and assist in maintaining uniform boron concentrations in the RCS and the refueling pool above the LCO limits. Administrative controls will limit the volume of unborated water that can be added to the refueling pool for decontamination activities in order to prevent diluting the refueling pool below the specified limits (Ref. 3).

APPLICABLE SAFETY ANALYSES

The boron concentration LCO limits are 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% $\Delta k/k$ margin of safety is established during refueling. Safety analyses assume a B-10 abundance of 19.9 atom % (Ref. 4). Administrative controls ensure that the reactivity insertion from the reactor coolant system and the refueling pool reflects this assumption.

During refueling, the water volume in the refueling pool and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes having direct access to the reactor vessel.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). Boron dilution accidents are precluded in MODE 6 by isolating potential dilution flow paths. See LCO 3.9.2, "Unborated Water Source Isolation Valves." Unacceptable dilution from refueling pool decontamination activities is precluded by the following (Ref. 3):

1. The maximum allowable amount of unborated reactor makeup water that may be added to the refueling pool for decontamination activities is calculated for each refueling and will not cause the refueling pool boron concentration to fall below the LCO limits. This maximum allowable volume is based on initial pool boron concentration and one-half the RCS volume at mid-loop.

2. The refueling pool is drained to approximately one foot above the reactor cavity seal/shield ring. The refueling pool is then drained via the reactor coolant drain tank pumps or other available means (excluding the RHR system) until the level is below the seal/shield ring. This directs potentially diluted water at the top of the pool away from the reactor vessel and core.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

3. After the level has been lowered to below the cavity seal/shield ring, further draining of the area enclosed by the inside diameter of the ring is performed via the RHR connection to the Chemical and Volume Control letdown line.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the filled portions of the RCS and the refueling pool, that have direct access to the reactor vessel while in MODE 6. The boron concentration limit ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations, and shall in all cases be ≥ 2000 ppm. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM), LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensure 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 stating that transition from MODE 5 to MODE 6 is not permitted while the LCO is not met. This Note specifies an exception to LCO 3.0.4 and prohibits the transition when boron concentration limits are not met. This Note assures that core reactivity is maintained within limits during fuel handling operations.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the filled portions of the RCS and the refueling pool that have direct access to the reactor vessel, is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES

ACTIONS (continued)

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the filled portions of the RCS and the refueling pool that have direct access to the reactor vessel, is within the LCO limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Chapter 15, Section 15.4.
3. Amendment 97 to Facility Operating License No. NPF-30, Callaway Unit 1, dated March 31, 1995.
4. Callaway Plant Request for Resolution 17070.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

Valves BGV0178 and BGV0601 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 pool and the fact that allunborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. FSAR, Section 15.4.6.
2. NUREG-0800, Section 15.4.6.
3. Amendment 97 to Facility Operating License No. NPF-30, Callaway Unit 1, dated March 31, 1995.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

There are two sets of source range neutron flux monitors;
(1) Westinghouse Source Range Neutron Flux Monitors, and
(2) Gamma-Metrics Source Range neutron flux monitors.

The Westinghouse source range neutron flux monitors (SE-NI-0031 and SE-NI-0032) are BF_3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1 to 1E+6 cps). The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1.

The Gamma-Metrics Source range monitors provide continuous visual indication in the control room to allow operators to monitor core flux.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide continuous indication to alert the operator to unexpected changes in core reactivity such as an improperly loaded fuel assembly.

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be operable, each monitor must provide visual indication in the control room.

The Westinghouse monitors are the normal source range monitors used during refueling activities. Gamma-Metrics Source Range Neutron Flux

(continued)

BASES

LCO (continued)

Monitor(s) are acceptable equivalent control room indication(s) for Westinghouse Source Range Neutron Flux Monitor(s) in MODE 6, including CORE ALTERATIONS, with the complete fuel assembly inventory set within the reactor vessel or with the Gamma Metrics Source Range Neutron Flux Monitor(s) coupled to the core. Reactor Engineering shall determine whether each monitor is coupled to the core.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. In other modes, the source range monitors are governed by LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, and LCO 3.3.9.

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The

(continued)

BASES

ACTIONS

B.2 (continued)

12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

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 Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. Neutron detectors are excluded from the CHANNEL CALIBRATION because it is impractical to set up a test that demonstrates and adjusts neutron detector response to known values of the parameter (neutron flux) that the channel monitors. Depending on which source range channels are used to satisfy the LCO, the Note applies to the source range proportional counters in the Westinghouse Nuclear Instrumentation System (NIS) or to the Gamma-Metrics fission chambers, as discussed in the Background and LCO sections above. The CHANNEL CALIBRATION of the Westinghouse NIS source range neutron flux channels consists of obtaining integral bias curves, evaluating those curves, and comparing the curves previous data. The 18 month Frequency is based on the need to obtain integral bias curves under the conditions that apply during a plant outage. The other remaining portions of the CHANNEL CALIBRATION may be performed either during a plant outage or during plant operation. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. FSAR, Section 15.4.6.

BASES

LCO (continued)

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the RCS temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling pool.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Pool Water Level." 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 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower

(continued)

BASES

ACTIONS

A.1 (continued)

boron concentration than that contained in the RCS because all unborated water sources are isolated and administrative controls are placed on refueling decontamination activities (See Bases for LCO 3.9.1).

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling pool water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition. Performance of Required Action A.2 shall not preclude completion of movement of a component to a safe condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level $\geq$ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

(continued)